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A Selected, Annotated Bibliography of Studies Relevant to the Isolation of Nuclear Wastes

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Information Division

A SELECTED, ANNOTATED BIBLIOGRAPHY OF STUDIES RELEVANT TO THE ISOLATION OF NUCLEAR WASTES

VOLUME 1

L. K. Hyder, C. S. Fore, N. D. Vaughan, and R. A. Faust

Ecological Sciences Information Center
Information Center Complex

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CONTENTS

HIGHLIGHTS v
INTRODUCTION vii
Measured Parameters viii
Citation Form viii
Services ix
Acknowledgments x
SAMPLE REFERENCE xi
BIBLIOGRAPHIC REFERENCES 1
Chapter 1. Chemical and Physical Aspects 1
Chapter 2. Container Design and Performance 39
Chapter 3. Disposal Site 65
Chapter 4. Environmental Transport 85
Chapter 5. General Studies and Reviews 123
Chapter 6. Geology, Hydrology and Site Resources 137
Chapter 7. Regulatory and Economic Aspects 195
Chapter 8. Repository Design and Engineering 207
Chapter 9. Transportation Technology 219
Chapter 10. Waste Production 221
Chapter 11. Waste Treatment 265
Indexes 273
Author 273
Keywords 283
Subject Category 297
Title 299
Geographic Location 357
Measured Parameters 361
Measured Radionuclides 363
Publication Description 365

HIGHLIGHTS

This annotated bibliography of 705 references represents the first in a series to be published by the Ecological Sciences Information Center (ESIC) containing scientific, technical, economic, and regulatory information relevant to nuclear waste isolation. Most references discuss deep geologic disposal, with fewer studies of deep seabed disposal; space disposal is also included. The publication covers both domestic and foreign literature for the period 1954 to 1980. Major chapters selected are Chemical and Physical Aspects; Container Design and Performance; Disposal Site; Environmental Transport; General Studies and Reviews; Geology, Hydrology and Site Resources; Regulatory and Economic Aspects; Repository Design and Engineering; Transportation Technology; Waste Production; and Waste Treatment. Specialized data fields have been incorporated to improve the ease and accuracy of locating pertinent references. Specific radionuclides for which data are presented are listed in the "Measured Radionuclides" field, and specific parameters which affect the migration of these radionuclides are presented in the "Measured Parameters" field. The references within each chapter are arranged alphabetically by leading author, corporate affiliation, or title of the document. When the author is not given, the corporate affiliation appears first. If these two levels of authorship are not given, the title of the document is used as the identifying level. Indexes are provided for (1) author(s), (2) keywords, (3) subject category, (4) title, (5) geographic location, (6) measured parameters, (7) measured radionuclides, and (8) publication description.

The bibliography was compiled from a specialized data base established and maintained by staff of the Ecological Sciences Information Center, Information Center Complex, Information Division, Oak Ridge National Laboratory, to provide information support for the Department of Energy's Office of Nuclear Waste Isolation (ONWI).

INTRODUCTION

This publication of 705 abstracted references represents the first in a series of annotated bibliographies to be published by the Ecological Sciences Information Center (ESIC), Information Center Complex, Information Division, Oak Ridge National Laboratory, relevant to the isolation of nuclear wastes. This compilation of literature is the result of efforts by ESIC to provide information support to the Department of Energy's Battelle Project Management Division, Office of Nuclear Waste Isolation, and its contractors.

The scope of the project focuses on deep geologic disposal in salt formations, shales, basalts, and granites. Also listed are studies of deep seabed, antarctic icesheet, and space disposal. Additional topics selected for inclusion in this publication deal with waste production and treatment, container design and performance, transportation of wastes, repository design, health and safety aspects, economics, and environmental and regulatory issues. The publication covers both domestic and foreign literature for the period 1954 to 1980. Major foreign papers were contributed by Sweden, Canada, and Germany. The bibliography was divided into the following eleven chapters to aid the reader in locating references of interest: Chemical and Physical Aspects; Container Design and Performance; Disposal Site; Environmental Transport; General Studies and Reviews; Geology, Hydrology and Site Resources; Regulatory and Economic Aspects; Repository Design and Engineering; Transportation Technology; Waste Production; and Waste Treatment. Information on social aspects of nuclear waste isolation has also been identified and collected by ESIC. This abstracted and indexed literature is stored in a separate data base and is available upon request. The present publication represents a first attempt at presenting a small portion of the existing literature collected by ESIC on the subject of nuclear waste isolation. Therefore, relevant publications not included in this issue will be abstracted and indexed for publication in future volumes.

The references to the literature are stored in a computer-searchable data file. Each reference in the data base contains a complete bibliographic citation, an informative abstract emphasizing important findings, subject categories describing the general subject area, keywords identifying specific topics in the document, and date and location of data-taking activities. Specialized data fields have been added to improve the ease and accuracy of identifying pertinent references. Specific radionuclides for which data are presented are listed in a field called "Measured Radionuclides." When information is presented on parameters which affect the migration of these radionuclides through the soil and geological formations, the parameters discussed are listed in a field entitled "Measured Parameters." The parameters are also discussed in the abstract along with numeric values, where possible. These fields enable the user to review only those references to documents concerned with the radionuclides and parameters of interest.

Measured Parameters

The following is a list of specific parameters that have been selected for inclusion in the Technical Aspects of Nuclear Waste Isolation Data Base. Data on these parameters are extracted from the literature and included in the abstract. These data are flagged in a separate field, labeled "Measured Parameters," to increase precision in locating pertinent references.

Activation Energy
Alkalinity
Aquifer Thickness
Bulk Modulus
Calcium Carbonate Content
Clay Content
Compressional Wave Velocity
Compressive Strength
Corrosion Rate
Creep Rate
Density
Depth to Water Table
Diffusion Coefficient
Distribution Coefficient
Eh
Elastic Modulus
Flow Rate
Grain Size Distribution
Heat Capacity
Heat of Vaporization
Hydraulic Conductivity
Hydraulic Gradient
Hydraulic Head
Hydraulic Velocity
Injection Pressure
Ion Exchange Capacity
Leaching Rate
Mass Action Coefficient
Mean Precipitation
Migration Rate
Moisture Content
Percent Adsorption
Permeability
Permeability Coefficient
pH
Plasticity
Poisson's Ratio
Porosity
Resistivity
River Flow Rate
Saturation Capacity
Shear Modulus
Sorption Coefficient
Specific Activity
Specific Conductance
Storage Coefficient
Stratigraphic Unit Thickness
Temperature
Tensile Strength
Thermal Conductivity
Thermal Expansion
Thermal Load
Total Dissolved Solids
Total Ion Concentration
Transmissivity
Viscosity
Volumetric Water Content
Waste Volume
Young's Modulus

Citation Form

The references within each chapter are arranged alphabetically by leading author, corporate affiliation, or title of the document. When the author is not given, the corporate affiliation will appear first. If these two levels of authorship are not given, the title of the document is used as the identifying level. The following indexes are provided to aid the reader in locating references of interest: (1) author(s),

(2) keywords, (3) subject category, (4) title, (5) geographic location, (6) measured parameters, (7) measured radionuclides, and (8) publication description.

As a result of computer limitations in indicating superscripts and subscripts in the standard manner, certain conventions have been established in this bibliography:

1. X sub t (X being a variable) means X_t or X subscript t.

2. In chemical compounds and elements, NaIO3 (for example) means $NAIO_3$.

3. 10(E+3) or X(E-3) (E denoting exponent) means 10^3 or X^{-3}.

4. For units of measurement, such as centimeters, meters, and feet, X3 means X^3.

Services

The Ecological Sciences Information Center provides a number of specialized services to governmental, industrial, academic, and foreign requestors. These services include identifying, acquiring, organizing, indexing, abstracting, processing, storing, and maintaining information. Customized literature searches, as well as copies of specific published documents referenced in the bibliography, are provided upon submission of requests. Annotated and indexed bibliographies are published whenever a particular need is shown. Specialized data bases can also be created if a specific subject area has not previously been covered.

ESIC provides service to a large population of researchers involved in the environmental concerns of energy technology. These services are made available at no charge to the project's sponsoring agency or agencies and to their contractors. Requests received from other researchers and decision makers will be assessed a minimum fee of $30, which covers the charges for most services. The fees will be charged through the National Technical Information Service in Springfield, Virginia. All inquiries for information services should be addressed to:

Ecological Sciences Information Center
Oak Ridge National Laboratory
P.O. Box X, Building 2001
Oak Ridge, Tennessee 37830

Phone: 615/574-7769 or
FTS 624-7769

Acknowledgments

Dr. Beverly Rawles of Battelle's Office of Nuclear Waste Isolation guided ESIC staff members in the selection of material for this project. Staff members of the Computer Services Group of the Information Sciences and Operations Department, Information Center Complex, were responsible for the computer production of this document.

SAMPLE REFERENCE

This is an example of the format for the descriptive fields used in this bibliography:

1 — *Chapter Heading*
2 — *Record Number (Sequential Number of Reference)*
3 — *Author(s)*
4 — *Corporate Affiliation*
5 — *Document Title*
6 — *Publication Description*
7 — *Publication Date*
8 — *Abstract*
9 — *Abstractor's Initials*
10 — *Comments*
11 — *Measured Parameters*
12 — *Measured Radionuclides*
13 — *Keywords*

[1]GEOLOGY, HYDROLOGY AND SITE RESOURCES

[2]<000>

[3]Droste, J.B. [4]University of Indiana, Bloomington, IN

[5]Paleozoic Stratigraphy of Two Areas in Southwestern Indiana.
[6]Y/OWI/SUB-706212; 53 pp. [7](1976, September)

[8]Two areas recommended for evaluation as solid waste disposal sites lie along the strike of Paleozoic rocks in southwestern Indiana (Martin, Crawford, and Perry Counties). Thin Pennsylvanian rocks of the upper Mississippian are at the bedrock surface in maturely dissected uplands in both areas. The gross subsurface stratigraphy beneath both areas is the same, but facies and thickness variation in some of the subsurface Paleozoic units provide for some minor differences. Thick Mississippian carbonates grade downward into clastics of lower Mississippian (Borden Group) and upper Devonian (New Albany Shale) rocks. Middle Devonian and Silurian rocks are dominated by carbonate lithologies. Upper Ordovician shales and carbonates (Maquoketa Group) overlie carbonates of Middle Ordovician age. Both areas will provide two rock zones with suitable geological and engineering properties to permit development of repository facilities for underground waste disposal. [9](RAF)

[10]Maps are given showing the thickness distribution and structures of geologic groups found in both northern and southern areas.

[11]Stratigraphic Unit Thickness

[12]Tc 99

[13]STRATIGRAPHY; WASTES, SOLID; WASTES, RADIOACTIVE; REPOSITORY; SITE SELECTION; GEOLOGY; WASTE DISPOSAL; BEDROCK; SHALES; SILTSTONES; SANDSTONES; LIMESTONES; GEOLOGIC STRUCTURES; OUTCROPS; GEOLOGIC HISTORY; WELLS; FIELD STUDIES

CHEMICAL AND PHYSICAL ASPECTS

<1>
Allard, B., H. Kipatsi, B. Torstenfelt, and J. Rydberg, Chalmers University of Technology, Department of Nuclear Chemistry, Fack, Gotebrg, Sweden.

Nuclide Transport by Groundwater in Swedish Bedrock.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 403-410), 563 pp. (1979)

By the year 1985 Sweden's nuclear power reactors will produce a minimum of 230 tons of used fuel per year and after reprocessing and solidification the corresponding amount of high-level waste, as glass, would be about 50 tons. As part of management program retention of long-lived fission products and actinides by natural clay and rock minerals has been studied. Recalculated surface dependent distribution coefficients were derived from sorption experiments on fresh rock surfaces and old crack surfaces. Sorption mechanisms have been suggested. Chemical equilibria for retention were investigated. The influence of the redox potential of the water on the formation of different species and changes in sorption due to changes in valence state have been studied. An estimation of the valence state and chemical form of the actinides U, Np, and Pu was been done in regards to the hydrolysis of the actinides and considering the expected pH-Eh conditions. Using the values derived from the above investigations relevant retention facts of the radionuclides have been proposed. (MDV)

Waste Volume

FISSION PRODUCTS; ACTINIDES; REACTORS; NUCLEAR POWER; SORPTION; DISTRIBUTION COEFFICIENT; HYDROLYSIS; WASTES, HIGH-LEVEL; GLASS; MINERALS; WASTES, RADIOACTIVE; LABORATORY STUDIES

<2>
Rees, L.L., Jr., Battelle-Pacific Northwest Laboratories, Richland, WA.

Linde 13X Equilibrium with Trivalent Cerium and Yttrium Cations.

BNWL-36 (Part 4); Hanford Radiological Sciences Research and Development Annual Report for 1964, D.W. Peace and J.K. Green (Eds.), (pp. 4.8-4.12), 571 pp. (1965, January)

Cation exchange equilibria of zeolite Linde 13X in a sodium-cerium, sodium-yttrium, and strontium-cerium systems were studied. It was found that the zeolite has an ion capacity for cerium of 3.6 meq/g. A selectivity coefficient was determined at equilibrium and corrected for nonideal behavior of the cations in the solution and on the zeolite. The method is outlined in the appendix. The zeolite is highly selective for cerium in the presence of sodium but prefers strontium to both cerium and sodium. (MDV)

Ion Exchange Capacity

CATIONS; CATION ADSORPTION EQUILIBRIUM CONSTANT; EQUATIONS; ION EXCHANGE CAPACITY; LABORATORY STUDIES; RARE EARTHS; ZEOLITES

<3>
Anders, O.U., The Dow Chemical Company Analytical Laboratories, Midland, MI.

Low-Level Solidified Waste Characterization: The Scientific Basis for Physical and Chemical Testing.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 499-500), 563 pp. (1979)

It is the purpose of this paper to present the initial accomplishments of an effort to provide the scientific base for burial site acceptance criteria for solidified radwastes. Since leachability of the waste form is the most significant characteristic, a standard is in preparation by the American Nuclear Socity. For burial site acceptance criteria, six guidelines are proposed: 1) disregard effects due to packaging of the radioactive ingot (in order to evaluate those factors separately); 2) the waste ingot must have a "sound life" expectancy exceeding 20 half-lives of the longest-lived radioactive constituent present in concentration exceeding the specific radioactivity of uranium ore; 3) the specific radioactivity shall not exceed that value which would by radiation damage reduce the "sound life" expectancy needed; 4) the release of radioactivity must remain below 5% of the initial radioactivity content of the waste ingot during any time interval of length 5% of the defined "sound life" of the material; 5) flammability and heat stability of the ingot small meet criteria aplied to construction materials of similar size and shape; 6) the waste ingot must retain its shape and dimensions so as not to cause collapse of the trench. (16th) (DCW)

WASTES, RADIOACTIVE; WASTE STORAGE; WASTE DISPOSAL; WASTES, SOLID; BURIAL; SOLIDIFICATION; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; RADIOACTIVITY; LEACHING; REPOSITORY; STANDARDS; REGULATIONS

<4>
Anderson, T.J., Savannah River Laboratory, Aiken, SC.

Methodology for the Determination of Environmental Iodine 129 and Technetium 99.

DP-MS-77-75X; 3 pp. (1978)

Analytical methods are being developed at the Savannah River Laboratory to determine the environmental impact of the Savannah River Plant with regard to I 129 and Tc 99. Both isotopes are long-lived byproducts of nuclear fuel reprocessing. Iodine 129 is being determined by a neutron activation procedure in which I 129 and I 127 are activated to I 130 and I 128, respectively. Both products are then quantified by Ge(Li) gamma ray spectroscopy after chemical isolation. Chemical yield is determined by measuring recovery of an I 125 spike. Iodine 129 concentrations as low as 10(-5) pCi/g in the soil can be determined with counting times of

<4>

CHEMICAL AND PHYSICAL ASPECTS

<4> CONT.
less than 30 min (10% rel. std. dev.). An isotope dilution method for Tc 99 employing a three stage surface ionization mass spectrometer with a 10(E+9) abundance sensitivity is being developed. The current concept of the isolation scheme based on preliminary experiments involves a combination of ion exchange and solvent extraction steps, ending with the Tc 99 loaded on a single ion exchange bed for mass spectrometry. Technetium 97 is added to determine the chemical yield of the separation procedure. A lower detection limit of about 10(E-18) g is sought. Both methods will be used to study I 129 and Tc 99 relative to 1) concentration in soil and vegetation as a function of distance from the SRP, 2) vertical migration from the soil surface, and 3) underground lateral transport from radioactive waste basins and storage sites. (Auth) (LKM)

Summary only is given.

I 129; Tc 99

IODINE 129; TECHNETIUM 99; SOIL TRANSPORT; RADIONUCLIDE MIGRATION; METHODS; INSTRUMENTS; CHEMICAL ANALYSIS; SPECTROMETRY; ION EXCHANGE; DISPOSAL SITE; LABORATORY STUDIES

<5>
Archambault, J., J. Lemaine, and V. Hugent (Translator).

Underground Storage Co.Jission for Radioactive Wastes.

ORNL/tr-4638; Retention and Migration of Radioactive Ions in Soils, Proceedings of an International Colloquium, Saclay, France, October 16-18, 1962, (pp. 21-23). (1963)

When storing radioactive wastes the prime concern is the possibility of contaminating water tables for either long or short periods of time. Wastes may be placed in a closed hydrological basin in an arid climate, treated to form either solid or insoluble substances or placed in water tight drums, injected or placed into deep formations, or fixed by the soil. Better solutions are storage in abandoned tunnels, injections into schistose series or placement into salt deposits. The tunnels, formerly used by trains, could be made water tight and if this is not possible then could be closely monitored. Waste may be implaced in the salt deposits in drums in galleries or as effluent directly emptied into man-made cavities. Two types of salt deposits exist, those with interstratified layers which are subject to dissolution, and massive domes (diapirs) which offer great safety because of their massiveness. (NDV)

BURIAL; CAVITIES; DRUMS; GEOLOGY; SALT DEPOSITS; DIAPIRS; SCHIST ; WASTE DISPOSAL; WASTE STORAGE; WASTES, RADIOACTIVE; HYDROLOGY; WELLS, INJECTION; SOLIDIFICATION; SOILS; DISSOLUTION; FIXATION; HAZARD ANALYSIS

<6>
Baes, C.F., Jr., Oak Ridge National Laboratory, Oak Ridge, TN.

Studies on the Hydrolysis Reactions of Metal Ions.

Y/OWI/SUB-77/14268; CONF-770749; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 23-24), 92 pp. (1977, August 20)

At the intermediate pH of groundwaters the species of most metallic ions produced in solutions at low concentrations are mononuclear hydrolysis products. The stability of these, especially the intermediate ones, is often poorly known because at the higher concentrations where most methods of such measurement are suitable, polynuclear hydrolysis products are the dominant species in solution and/or the pH range is limited by hydrolytic precipitation. Perhaps the best method for studying the mononuclear species is to measure the solubility of the oxide or hydroxide solid phase that is stable under the conditions of interest. Many such studies that have been attempted in the past have been compromised because the solid used was not a well characterized crystalline phase of sufficient particle size and/or because the ionic strength was not controlled to minimize activity coefficient and equilibrium quotient variation. Currently, column solubility measurements of gibbsite, Al(OH)3, are being conducted in NaCl solution as a function of pH, temperature, and ionic strength in order to better establish the stability of the intermediate species Al(OH)2+ and Al(OH)3(aq). Studies are planned for ThO2, representative of the tetravalent actinides, and of structural metal oxides and hydroxides. (Auth) (LKM)

Only the abstract is given.

pH; GROUND WATER; HYDROLYSIS; IONS; IONIC PROCESSES; OXIDES; HYDROXIDES; SOLUBILITY; CONCENTRATION; METALS; ACTINIDES; THORIUM; LABORATORY STUDIES; CHEMICAL PROPERTIES; CHEMICAL ANALYSIS

<7>
Barnhart, B.J., E.W. Campbell, J.W. Hardin, E. Martinez, D.E. Caldwell, and R. Mallett, Los Alamos Scientific Laboratory, Los Alamos, NM.

Potential Microbial Impact on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP). December 15, 1978-March 15, 1979.

LA-7839-PR; 11 pp. (1979, July)

This report discusses work conducted in the second quarter of FY 1979 relating to potential microbial impacts on transuranic wastes under conditions expected at the Waste Isolation Pilot Plant (WIPP). Soil samples were taken from depths of 5-15 cm, 40-50 cm, and 100-110 cm in a trench at LASL burial site TA-54, Area C. Soil radioactivity for these samples was not detectable at 5-15 cm, was 4 X 10(E+4) dpm gamma, 6 X 10(E+4) dpm beta at 40-50 cm, and was 2 X 10(E+3) dpm gamma, 7 X 10(E+3) dpm beta at 100-110 cm. The number of microflora in these samples decreased with depth. Although the level of radioactivity was greatest in the 40-50 cm sample, the frequency of microorganisms was intermediate between that of the other depths, indicating these levels of radioactivity apparently do not exert any

<7> CONT.
sterilizing effect on the soil. The microflora on flammable (organic) waste contaminated with Pu 239 were enumerated and bacteria were classed as bacilli or cocci and by reaction to gram stain. Most probable numbers (MPN) for aerobic and fungal cultures from these materials were as follows: wood chips = 70 MPN aerobics (gram + spore-forming rods), no detectable fungi; Kleenex box (inner panel) = 110 MPN aerobics (gram + spore-forming rods), no detectable fungi; PVC glove (2 fingers) = 9.5 X 10(E+3) MPN aerobics (gram + cocci), 46 MPN fungi; tissue (1) = 33 MPN aerobics (gram + rods), 20 MPN fungi; cotton gauze (2 cm sq) = 23 MPN aerobics (not identified), no detectable fungi. The anaerobic cultures were contaminated, and so discarded. Microflora were also enumerated on a sample of LASL-simulated organic waste. Soil samples from 5-15 cm, 40-50 cm, and 100-110 cm were cultured and isolated bacterial and actinomycete colonies subjected to doses of 0, 13.5, 27, 40.5, and 54 krads of 250 kVP x-rays at 900 rads/min. Evaluation of the 5-15 cm isolates and the 40-50 cm isolates showed 3 resistant phenotypes out of 12 isolates for the former (25% resistant) and 13 out of 19 (68%) resistant phenotypes for the latter sample depth. Isolates with resistant phenotypes have been stored at -80 C for future biochemical and genetic characterization. The alkylation of Eu and Th using methylcobalamine as a methyl donor was attempted but no organometallic product other than cobalt-cobalamine could be detected chemically or by mass spectrometry. A statistical analysis of the rates of degradation of europium tartarate and europium citrate chelates is presented. (Auth) (LKH)

PLUTONIUM 239; BACTERIA; WASTES, ORGANIC; WIPP; REPOSITORY; MICROORGANISMS; CHELATES; COMPLEXES; TRANSURANICS; SOILS; DISPOSAL SITE; WASTES, TRANSURANIC; RADIATION EFFECTS; DEGRADATION; LABORATORY STUDIES; EUROPIUM; THORIUM; WASTES, RADIOACTIVE; METHODS; CHEMICAL REACTIONS

<8>
Barnhart, B.J., E.W. Campbell, J.H. Hardin, E. Martinez, D.E. Caldwell, and R.E. Hallett, Los Alamos Scientific Laboratory, Los Alamos, NM.

Potential Microbial Impact on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP). October 1-December 15, 1978.

LA-7788-PR; 17 pp. (1979, May)

This is the second quarterly progress report for this project designed to assess potential microbial interactions with plutonium and other transuranics in organic matrix radioactive wastes as may possibly occur in the Waste Isolation Pilot Plant (WIPP). Since October 1, microorganisms in soil from an area adjacent to a shallow burial site (TA-54, Area C) at the Los Alamos Scientific Laboratory (LASL) have been enumerated. From 1.32 X 10(E+6) to 2.98 X 10(E+6) aerobic colony forming units (CFU) were found per gram of soil as compared with 1.99 X 10(E+3) to 2.97 X 10(E+3) anaerobic CFU per gram. The fungus population ranged from 3.5 X 10(E+3) to 3.2 X 10(E+4) CFU per gram. The most probable number (MPN) demonstrated in nutrient broth by the extinction dilution method was similar for aerobic and anaerobic bacteria [2.3 X 10(E+6) for aerobic, 3.3 X 10(E+6) for anaerobic]. Direct counting of microflora in fluorescent-stained soil samples has been initiated using the epifluorescent microscopy capability at the University of New Mexico. The almost quantitative methylation of mercury has been demonstrated and an attempt has been made to alkylate europium in an abiotic reaction with methylcobalamine, a metabolic intermediate in bacterial metabolism, but methylation was not detected over a range of pH (4.0-7.0) and at two temperatures (25 C and 37 C). The alkylation of thorium is being pursued as a prerequisite to working with plutonium. Studies of europium citrate and europium EDTA chelate degradation have begun. Both degrade, but EDTA degrades at a much slower rate than citrate. Methodology has been improved in preparation for analogous studies of plutonium chelate degradation. The microbial production of gases under conditions expected in the WIPP has been studied. Carbon dioxide is the only significant gas detected. Hydrogen sulfide is also produced if a high concentration of proteinaceous substrate is present. (Auth) (LKH)

PLUTONIUM; TRANSURANICS; WASTES, ORGANIC; WASTES, TRANSURANIC; WASTES, RADIOACTIVE; WIPP; DISPOSAL SITE; BACTERIA; MICROORGANISMS; SOILS; METHODS; MERCURY; EUROPIUM; COMPLEXES; THORIUM; CHEMICAL REACTIONS; CHELATES; EDTA; DEGRADATION; GASES; LABORATORY STUDIES; RADIATION EFFECTS

<9>
Barnhart, B.J., E.W. Campbell, E. Martinez, D.E. Caldwell, and R. Hallett, Los Alamos Scientific Laboratory, Los Alamos, NM.

Potential Microbial Impact on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), March 15-June 15, 1979.

LA-7918-PR; 14 pp. (1979, July)

This report discusses work conducted in the third quarter of FY 1979 relating to potential microbial impacts on transuranic wastes under conditions expected at the Waste Isolation Pilot Plant (WIPP). It was shown, for the first time, that the chelate Desferol, which is representative of microbially produced polyhydroxamate chelates, greatly enhances the dissolution of plutonium from insoluble Pu 239 PuO_2. Radiobiological studies showed that bacterial isolates from Los Alamos Scientific Laboratory transuranic burial site soil exhibit levels of gamma radiation resistance intermediate to that of BACILLUS SUBTILUS and MICROCOCCUS RADIODURANS. Also included in this report are the first tables of data from the CO_2 gas generation experiments with the corresponding modeling equations derived from the linear regression statistical analysis procedure. Data for the chelate degradation studies are updated and include data for EDTA chelate. Results of the experiment to enumerate and identify microorganisms in the salt beds of the WIPP facility are reported as negative, as no microbial growth was detected from three salt samples. (Auth) (LKH)

Pu 239

WIPP; REPOSITORY; BACTERIA; MICROORGANISMS; CHELATES; PLUTONIUM; TRANSURANICS; DISSOLUTION; RADIONUCLIDE MIGRATION; SOILS; DISPOSAL SITE; WASTES, TRANSURANIC; RADIATION EFFECTS; GASES;

<9>

CHEMICAL AND PHYSICAL ASPECTS

<9> CONT.
DEGRADATION; SALT DEPOSITS; SALTS; LABORATORY STUDIES

<10>
Bind, G.H., and J.V. Biggers, The Pennsylvania State University, University Park, PA.

Deformation Mapping of Polycrystalline Salt.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978, (pp. 65-66), 77 pp.(1978)

An important consideration in repository design is the deformation behavior of polycrystalline salt as a function of stress, temperature and time. Presented are extended ideas on the modelling of creep behavior and the use of deformation maps. Possible creep mechanisms that may operate in NaCl are: defectless flow, deformation by dislocation glide, power law creep, diffusion flow, grain boundary diffusion, volume diffusion, and sized kinetics. For the initial modelling dislocation glide, power law creep, and atomic diffusion were considered most important mechanisms. Equations describing these processes in terms of material parameters, temperature, and stress were developed, and a computer solution using Newton's iteration made it possible to construct deformation maps. From a large number of these maps generated over a variety of grain sizes, the drastic effect of salt microstructure on creep behavior of low stress levels and temperatures (burial site conditions) is illustrated. Contours of constant strain can be used to predict repository stability over extended periods of time. (NDV)

This paper was presented at the conference but was not included in the Proceedings.

COMPUTER CODES; CREEP; STRESS; STRAIN; DEFORMATION; SALT DEPOSITS; REPOSITORY; MODELS, MATHEMATICAL; TEMPERATURE

<11>
Bondietti, E.A., and C.W. Francis, Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Geologic Migration Potentials of Technetium 99 and Neptunium 237.

Science 203(4387):1337-1340, (1979, March 30)

Research at Oak Ridge National Laboratory indicates that current risk assessments which consider Tc and Np as potentially capable of migrating from high-level radioactive waste repositories may be overestimating their hazard to the public, since the Fe(+2) content of many groundwaters may maintain these elements in less soluble oxidation states. Previous studies which found Tc to be poorly sorbed by geologic media used the pertechnetate oxyanion TcO_4^- under oxidizing conditions, while conclusions that Np is a relatively mobile element compared to Pu were based on data obtained with the oxycation NpO_2^+, also under oxidizing conditions. In this study, comparison of calculated values for $TcO_2(s)$ and $NpO_2^+/NpO_2(s)$ Eh-pH equilibrium lines were compared with measurements of Eh values for samples of weathered and unweathered Conasauga Shale, an argillaceous shale, pH-adjusted and normal Westerly Granite, Climax Stock granite, and pH-adjusted and normal Columbia Cap Basalt, in deoxygenated water. In a standard Eh-pH diagram, the three igneous rocks exhibited Eh values below the Tc line but above the hematite/magnetite line; shales on the other hand exhibited Eh values above the Tc line but below the Np line. These results indicated that TcO_4^- should be unstable with respect to $TcO_2(s)$ in the presence of the igneous rocks. This was verified experimentally when 99% of the Tc 99 added to a stirred cell containing one of the rocks was lost from solution in less than 3 days. A slow but significant rate of Tc removal by the weathered Conasauga sample was also noted. Neither the subsurface Conasauga nor the argillaceous shales removed Tc (concentration 0.11 uM) over a 7 day period. Other experiments confirm that TcO_4^- is not stable when equilibrated in the absence of air and that Tc is reduced even at low concentrations [less than 8 X 10(E-12) M]. Experiments showed rapid removal of Np [initially Np(+5)] from solution during equilibrations. The shales, because of their greater surface area, removed Np(+5) faster than the igneous rocks. It was shown that with the basalts and granites, $Np(+4)$ dominated the solution-phase species after 7 days of equilibration; values ranged from 66-71%. Np associated with the rocks was also predominantly Np(+4) (75-82%). Oxidized basalt (pretreated with NaOCl) showed no reaction. The unweathered shales were not active in reducing Np(+5). (LKH)

Eh; pH

Np 237; Tc 99

TECHNETIUM; NEPTUNIUM; VALENCE; IGNEOUS ROCKS; REDOX REACTIONS; HAZARD ANALYSIS; WASTES, HIGH-LEVEL; REPOSITORY; GROUND WATER; STORAGE, GEOLOGIC; RADIONUCLIDE MIGRATION; LABORATORY STUDIES; SHALES; Eh; pH; GRANITES; BASALTS; SEDIMENTARY ROCKS

<12>
Bradshaw, R.L., Oak Ridge National Laboratory, Health Physics Division, Oak Ridge, TN.

Field and Laboratory Studies Preliminary to Main Experiment.

ORNL-4555 (Draft); ORNL-CF-70-7-82; Project Salt Vault: A Demonstration of High Activity Solidified Wastes in Underground Salt Mines, Ch. 4, (pp. 4.1-4.72), 822 pp.(1971, April)

The chapter contains discussions on the studies done on plastic flow, deformation rates, recovery of thermal expansion, stress tests, creep, corrosion, radiolytic chlorine production, and dosimeters. Detailed conclusions are given. (CAS)

Schematic drawings, graphs and tables are included.

WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; GEOPHYSICAL SURVEYS; DISPOSAL SITE; DESIGN; THERMAL PROPERTIES; MINES; LABORATORY STUDIES; RADIATION EFFECTS; SITE EVALUATION; CHLORINE; DOSIMETERS; PLASTICITY; STRESS;

<12> CONT.
CORROSION; DEFORMATION; WASTES, SOLID; FIELD STUDIES

<13>
Bradshaw, W.L., Oak Ridge National Laboratory, Health Physics Division, Oak Ridge, TN.

Summary of Previous Pertinent Studies.

ORNL-4555 (Draft); OPWL-CF-76-7-42; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 2, (pp. 2.1-2.26), 822 pp. (1971, April)

The thermal stability of salt under elevated temperatures due to fission product decay is studied. Results of the studies indicate that effects on trapped moisture, creep, structural properties, mine stability, shielding properties, and heat transfer properties are neglible. Cost estimates showed an economic competitiveness with nuclear power. (CAB)

Several graphs, tables and photos are included.

WASTE MANAGEMENT; WASTES, SOLID; WASTES, RADIOACTIVE; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; GEOPHYSICAL SURVEYS; RECOMMENDATIONS; THERMAL PROPERTIES; MINES; LABORATORY STUDIES; THEORETICAL STUDIES; ECONOMICS; MOISTURE

<14>
Brindley, G.W., Pennsylvania State University, Department of Geosciences and Materials Research Laboratory, University Park, PA.

Sorption and Fixation of Large Cations by Shale Formations.

Y/OWI/SUB-77/18268; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 33-34), 92 pp. (1977, August 20)

Large cations such as Cs+ are preferentially sorbed and partially fixed by fine-grained 2:1 type layer silicates such as micas, vermiculites, smectites and various mixed-layer minerals. Since these minerals are common constituents of many shale formations, these formations may provide suitable locations for the burial of radioactive waste containers. However, the usual investigations of cation sorption and fixation must be extended to the conditions likely to develop in the vicinity of buried radioactive waste. Prior to possible leakage from buried containers, elevated temperatures around the burial site will develop, leading to hydrothermal modification of the surrounding mineralogy. A range of temperature and of mineralogy must be envisaged. If and when leakage occurs, an outward diffusion of radioactive ions will occur by solvation in the fluids in the shale. The ratio of radioactive ions/normal ions will diminish outwards from the source. At near distances from the source high temperature modifications of the clay minerals and high concentrations of radioactive cations may lead to saturation of the fixation capacity. At greater distances, little or no thermal modification of the clay minerals and lower concentrations of ions will permit maximum sorption and fixation. The minerals studied to date include 1 phlogopite, 4 vermiculites, and 2 illites. The procedure was treatment four times with 0.001 M CsCl solution and once with 0.05 M CsCl. The amount sorbed was then measured by atomic absorption methods. Fixation was determined using a solution mixed with NaNO3 at Cs/Na = 1/600. Cesium was fixed at greater than 1/600, indicating a strong preference for Cs; this may be explained by the lower hydration energy of Cs versus Na. Samples have also been leached with strong KCl solutions in an effort to move the Cs. (Auth) (LKL)

Only the abstract is given.

SHALES; SEDIMENTARY ROCKS; CESIUM; SILICATES; CLAYS; MICA; VERMICULITE; SMECTITE; SORPTION; FIXATION; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; THERMAL PROPERTIES; HYDROTHERMAL ALTERATIONS; RADIONUCLIDE MIGRATION; ILLITE; PHLOGOPITE; LABORATORY STUDIES; WASTE-ROCK INTERACTIONS;

<15>
Brookins, D.G.

Geochemical Constraints on Accumulation of Actinide Critical Masses from Stored Nuclear Waste in Natural Rock Repositories.

ONWI-17; 52 pp. (1978, December 31)

This report presents the results of a literature search of data available on lanthanide and actinide individual and joint systematics. Covered were selected papers and reports about uranium solution chemistry, uranium deposits, the natural fission reactor at Oklo, West Africa, rare-earth deposits, manganese nodules, bedded and dome salt deposits, and miscellaneous items. Approximately 800 articles were inspected; 69 are included in the References Cited. The data obtained indicated that only limited segregation of the actinides U, Np, Pu, Am, and Cm from the lanthanides is possible should high-level waste be released from canisters stored in various geomedia. Supporting this were studies of Oklo and other uranium deposits, manganese nodules, monomineralic and concretion formation rates, and actinide and lanthanide transport in brines. The fact that some waste canisters may, under certain conditions, contain several critical masses of one or more actinides is countered by the facts that (a) most actinides have very short half-lives and would decay before release from canisters, (b) released actinides and lanthanides, although dispersed, would be transported and deposited as a group, thus preventing point concentration of any actinides, and (c) U 235 has a much longer half-life than the other actinides, thus allowing greater time for possible reaccumulation and criticality; such a scenario would demand that U 235 be effectively segregated from other elements in the lanthanide-actinide groups. No mechanism to do this is consistent with the natural occurrences studied or the theoretical Eh-pH diagrams considered. (Auth) (LKM)

U; Np; Pu; Am; Cm

ACTINIDES; LANTHANIDES; REVIEWS; GEOCHEMISTRY; THEORETICAL STUDIES; WASTES, HIGH-LEVEL; WASTE

<15>

CHEMICAL AND PHYSICAL ASPECTS

<15> CONT.
DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; RADIONUCLIDE MIGRATION; Eh - pH; URANIUM; NEPTUNIUM; PLUTONIUM; AMERICIUM; CURIUM; REACTORS, NATURAL; GEOLOGIC DEPOSITS; BRINES; SALT DEPOSITS; SALT DOMES; GRANITES; IGNEOUS ROCKS; METAMORPHIC ROCKS; CHEMICAL PROPERTIES; HALF-LIFE, RADIOLOGICAL

<16>
Clynne, M.A., and R.W. Potter, II, U.S. Geological Survey, Menlo Park, CA.

P-T-X Relations of Anhydrite and Brine and Their Implications for the Suitability of Anhydrite as a Nuclear Waste Repository Medium.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 323-328), 563 pp. (1979)

A major disadvantage of halite as a repository medium for high-level radioactive wastes is its high solubility and positive thermal coefficient. (As heat increases, included brines in the halite migrate toward the heat.) Anhydrite ($CaSO_4$) is considered a possible repository medium because of its sparing solubility in water and its negative thermal gradient. However, in chloride brines its solubility increases and its thermal coefficient changes. To test the combined effects of chloride ions, temperature, and pressure on the properties of anhydrite, the solubility of anhydrite in concentrated brines was studied using NaCl-saturated water, and a bittern brine (NBT-6) with 30 wt % total dissolved solids of Na, K, Ca, and Mg chlorides in a ratio of 1:1:2:2 on a weight basis. A static variation of an indirect method of determining salt solubilities in aqueous solutions at elevated temperatures was used, and $CaSO_4$ solubilities were determined from a series of precise vapor pressure measurements of $CaSO_4$ solutions. $CaSO_4$-saturated vapor pressure curves for NaCl-saturated and NBT-6 solutions were established, and measurements were taken of $CaSO_4$ solubility in the NaCl-saturated solution and the NBT-6 solution. Though comparison of the results with previous experimental data was difficult, extrapolation of three data sets showed favorable comparisons with this study. The solubility properties of anhydrite indicate that it could [illegible] as a suitable repository medium. The solubility of $CaSO_4$ remains low despite high chloride ion activity. At 200 C the solubility of anhydrite in saturated NaCl solutions is approximately 3.9 X 10(E+2) times less soluble than NaCl. The solubility of anhydrite in NaCl-saturated water exhibits a negative temperature coefficient up to 235 C (+ or - 20 C). We have shown that the solubility of anhydrite in NBT-6 is significantly less than in NaCl-saturated water, and that its negative temperature coefficient persists to higher temperatures (250 C + or - 20 C). Consequently, fluid inclusions will migrate down a thermal gradient, away from radioactive waste containers. The suitability of anhydrite as a repository medium is further enhanced by its thermal conductivity, its favorabble occurrence in evaporite basins, and its generally low fluid inclusion content. (Auth) (DCH)

Temperature; Pressure (Vapor); Solubility (of $CaSO_4$ in two brine solutions)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTE STORAGE; WASTE DISPOSAL; ANHYDRITE; HALITE; SALT DEPOSITS; EVAPORITES; SULFATES; MINERALOGY, GEOLOGIC DEPOSITS; BRINES; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; CHEMISTRY; GEOCHEMISTRY; REPOSITORY; LABORATORY STUDIES; METHODS; MEASUREMENTS; EXPERIMENTS; PRESSURE EFFECTS; SOLUBILITY; IONS; FLUID INCLUSIONS; THERMAL PROPERTIES; THERMAL CONDUCTIVITY; TEMPERATURE

<17>
Cohen, J.L., and T.L. Steinborn, Lawrence Livermore Laboratory, Livermore, CA.

The Rock-Melt Approach to Nuclear Waste Disposal in Geologic Media.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 261-264), 563 pp. (1979)

Decay heat has been considered a problem for radioactive waste disposal in geologic media, both from the standpoint of degradation of the waste itself and the alteration of the hydrogeologic environment such that groundwater could gain access to it. The waste heat could be beneficial to the unconventional rock-melt concept of disposal, in which a high-level waste solution or slurry is pumped into a deep mined cavity in dry silicate rock. The cavity boils dry and the decay heat melts the waste and rock together over tens to hundreds of years. Over several hundred more years, the melt cools, encapsulating whatever radioactive products are left in a insoluble rock matrix deep underground. In siting such a facility, no pre-disposal processing of the high-level waste solution would be necessary, since solidification is not required. To eliminate waste transport problems, co-location of the fuel reprocessing plant and waste repository would be advantageous. Rock types to look for could be a dry granite body or thick, dry sediments over the water table. The depth of burial and the host rock chemistry could be selected so that the diluted waste would be in slowly cooled mineral phases extremely resistant to leaching (such as monazite, zircon, thorite, etc.), or the "synrock" approach could be used. (DCH)

One table is provided which shows characteristics of some high-level radioactive wastes, including the product form, the process to arrive at the product, and water leachability.

WASTE STORAGE; WASTE DISPOSAL; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTE PROCESSING; WASTE-ROCK INTERACTIONS; SITE SELECTION; GEOLOGY; HYDROLOGY; GROUND WATER; SILICATES; SEDIMENTARY ROCKS; HOST ROCKS; CAVITIES; DECAY HEAT; RADIOACTIVE DECAY; THERMAL PROPERTIES; MONAZITE; STABILIZATION, CHEMICAL; STABILIZATION, PHYSICAL; METHODS; IMMOBILIZATION; ENCAPSULATION

<18>
Cook, N.G.W., Lawrence Berkeley Laboratory,

<18> CONT.
Earth Sciences Division, Berkeley, CA; University of California, Department of Materials Science and Mineral Engineering, Berkeley, CA.

An Appraisal of Hard Rock for Potential Underground Repositories of Radioactive Wastes.

LBL-7073; SAC-10; Mechanical and Thermal Design Considerations for Radioactive Waste Repositories in Hard Rock, (pp. 1-15), 26 pp. (1978, October)

Underground burial of radioactive wastes in hard rock may be an effective and safe means of isolating them from the environment and from man. The mechanical safety and stability of such an underground repository depends largely on the virgin state of stress in the rock, groundwater pressures, the strengths of the rocks, heating by the decay of the radioactive wastes, and the layout of the excavations and the disposition of waste canisters within them. A large body of pertinent data exists in the literature, and each of these factors has been analyzed in the light of this information. The results indicate that there are no fundamental geological nor mechanical reasons why repositories capable of storing radioactive wastes should not be excavated at suitable sites in hard rock. However, specific tests to determine the mechanical and thermal properties of the rocks at a site would be needed to provide the data for the engineering design of a repository. Also, little experience exists of the effects on underground excavations of thermal loads, so that this aspect requires theoretical study and experimental validation. The depths of these potential repositories would lie in the range from 0.5 km to 2.0 km below surface, depending upon the strength of the rock. Virgin states of stress have been measured at such depths which would retard the ingress of groundwater and obviate the incidence of faulting. A typical repository comprising three horizons each with a total area of 5 sq km would have the capacity to store wastes with thermal output of 240 MW. (Auth)

ROCKS; REPOSITORY; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; BURIAL; SAFETY; STRESS; GROUND WATER; STRENGTH, MECHANICAL; THERMAL PROPERTIES; HEAT TRANSFER; EXCAVATION; CONTAINERS; SITE SELECTION; ROCK MECHANICS; DESIGN; FAULTS; THEORETICAL STUDIES; EQUATIONS; REVIEWS; WASTES, HIGH-LEVEL

<19>
Cook, N.G.W., and P.A. Witherspoon, Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA; University of California, Department of Materials Science and Mineral Engineering, Berkeley, CA.

Mechanical and Thermal Design Considerations for Radioactive Waste Repositories in Hard Rock. Parts 1 and 2.

LBL-7073; SAC-10; 26 pp. (1978, October)

The two papers contained in this report discuss some of the fundamental considerations of the thermomechanical aspects of repository design. Although fundamental in scope, both papers also contain specifics which have significantly influenced the design of the heater experiments at Stripa. The conclusion of the first paper, "An Appraisal of Hard Rock for Potential Underground Repositories of Radioactive Waste", suggests that there are no fundamental reasons of a geomechanical nature why hard rock should not form a potentially satisfactory site for an underground repository of radioactive wastes. The second paper, "In Situ Heating Experiments in Hard Rock: Their Objectives and Design", relates the development of time-scaled heating experiments. Both papers are abstracted separately for the data base. (BAP)

REPOSITORY; THERMAL PROPERTIES; THEORETICAL STUDIES; FIELD STUDIES; HEAT SOURCES; INSTRUMENTS; ROCKS; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; ROCK MECHANICS; SITE SELECTION; TIME FACTOR

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Crouch, S.L., and W.C. McClain, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Interim Report on Development of a Semi-Empirical Numerical Model for Simulating the Deformation Behavior of a High-Level Radioactive Waste Repository in Bedded Salt.

ORNL/TM-5962; 80 pp. (1978, April)

The model assumes that the rock mass is isotropic and linearly elastic. It also assumes that all nonlinear, inelastic effects are confined to a single seam, or salt bed, where deformations are related to time, stress, and temperature. The seam thickness is assumed to be negligible in relation to the depth and lateral extent of the mined areas; the mining horizon is viewed as a single plane surface, with relative movements (i.e., closures) between the two sides of the plane used to specify the state of deformation and stress in the rock mass. Because of the assumed linear behavior of the rock, superposition is used to analyze the repository problem in terms of three interrelated phenomena: (1) elastic closure of the excavated areas with consequent elastic deformation of the rock mass; (2) time-, stress-, and temperature-dependent closure of the mined portions of the seam; and (3) thermal stress and displacement changes throughout the rock mass, including the seam, where stress and temperature variations affect the creep behavior of the seam material. The computer model is based on the displacement discontinuity method of analysis, in which the seam is treated as a number of individual displacement discontinuities (or dislocations) whose values are determined such that the appropriate boundary conditions are satisfied at the plane of the repository. The boundary conditions, in turn, are governed by the overburden stress, the thermal stress, and temperature changes caused by the decay of the radioactive wastes and the inelastic behavior of the salt in the seam. The radioactive waste is modeled as a series of heat sources with specified heat generation characteristics. The creep deformation of the seam is computed from a creep law whose form was derived from laboratory model pillar tests; the parameters for the creep law were established from full-scale field data obtained from the Project Salt Vault experiment. The computer program that was written to analyze the repository problem is briefly described, and the modifications

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CHEMICAL AND PHYSICAL ASPECTS

<20> CONT.

necessary to make it applicable to actual repository design studies are discussed. The current version of the Repository Program is based on a 50 by 50 grid of square elements (displacement discontinuities) representing the plane of the repository. The fundamental objective of the program is to find the closure occurring at each element as a function of time, because these values specify the state of deformation and stress in the surrounding rock mass. In order to permit the analysis of a full-scale repository, about 10,000 ft long by 5000 ft wide (and about 2000 ft deep), it is necessary to ignore the detailed room and pillar geometry and to treat the plane of the repository as a series of elements that are partially extracted, with some specified initial extraction ratio, and backfilled with crushed salt. As each element is "excavated," it is considered to act as a heat source with prescribed heat generation characteristics; these characteristics are determined by the density of radioactive waste canisters in the actual problem and are chosen such that the average heat generation per unit area is the same in the model as in reality. A series of check-out runs was performed to confirm that the various portions of the Repository Program were functioning satisfactorily and to investigate the sensitivity of the model to variations of particular parameters over appropriate ranges to confirm that the computer results were reasonable. These runs were performed successfully, and it is concluded that the Repository Program should be put into a form that is convenient for actual design studies. (Auth) (RAP)

WASTES, HIGH-LEVEL; THEORETICAL STUDIES; REPOSITORY; DESIGN; SALT DEPOSITS; MODELS, MATHEMATICAL; DEFORMATION; ROCK MECHANICS; COMPUTER PROGRAMS; ROCKS; ISOTROPY; MINING; EXCAVATION; STRESS; ELASTOPLASTICITY; CREEP; THERMAL PROPERTIES; HEAT SOURCES; CONTAINERS; EQUATIONS

<21>
Dames and Moore, White Plains, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Baseline Rock Properties--Granite.

Y/OWI/TM-36/5; 157 pp. (1978, April)

This report is an evaluation of the rock mass properties of granite for use in feasibility studies of geologic storage of radioactive wastes. Following a brief review of the available literature, four granites representative of different geologic ages and from different parts of the U.S. were selected for more detailed study. These were the Barre, Colville, Pikes Peak, and Charcoal granites. A table of intact properties for a typical granite was drawn up from the data; some of the special characteristics of granite, such as anisotropy, creep, and weathering were also investigated. Intact properties ranged as follows: Unit Weight 144-190 lbs/cu ft (mean 165.2); Natural Moisture Content 0-0.32%; Porosity 0.05-11.2% (mean 1.6%); Young's Modulus 2.3 x 10(E+6)-12.1 x 10(E+6) psi (mean 7.3 x 10(E+6)); Poisson's Ratio 0.065-0.39 (mean 0.18); Uniaxial Compressive Strength 5100-51,200 psi (mean 25,290); Tensile Strength 500-8100 psi (mean 910); Coefficient of Linear Thermal Expansion 3.0 x 10(E-6)-6.0 x 10(E-6)/deg F (mean 4.5 x 10(E-6)); Heat Capacity 0.16-0.33 BTU/lb-F (mean 0.22); typical Thermal Conductivity values are 1.65 BTU/hr-ftF at 32 F and 1.48 BTU/hr-ftF at 212 F. Since very few field tests and measurements of large-scale field behavior have been made, a considerable amount of engineering judgment was necessary in arriving at these rock mass properties. Given the present state-of-the-art and the available field information these properties are not unique but are considered to be reasonable. Before a repository is designed and built, a major program of laboratory and field testing will be necessary to determine the necessary rock mass properties for that particular site. (Auth)(LEH)

Volumetric Water Content; Porosity; Young's Modulus; Poisson's Ratio; Compressive Strength; Tensile Strength; Thermal Expansion; Heat Capacity; Thermal Conductivity

REVIEWS; GRANITES; PHYSICAL PROPERTIES; THERMAL PROPERTIES; REPOSITORY; SITE SELECTION; ANISOTROPY; IGNEOUS ROCKS; ROCKS; STORAGE, GEOLOGIC; ROCK MECHANICS

<22>
Dames and Moore, White Plains, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 6. Baseline Rock Properties - Shale.

Y/OWI/TM-36/6; 183 pp. (1978, April)

This report is a result of a literature survey of the rock properties of shales occurring in the United States from the stand point of suitability for geologic storage of radioactive wastes. Firstly, data was collected from a wide variety of sources in order to obtain insight for the range of properties encountered. Secondly, shales from four different parts of the United States were considered in detail, and used to produce a table of intact shale properties. Lastly, excavation problems in shale formations were reviewed and three tunnel jobs were written up as case histories. Owing to the wide variability in lithology and properties of shales occurring in the United States, it became necessary to focus the study on consolidated illite shales, as the type most nearly suitable for a waste repository. Using the information generated, a set of intact properties for a typical, consolidated illite shale was obtained. Correction factors, largely based on geological considerations, were then applied to the intact data in order to yield typical rock mass properties for this type of shale. These properties ranged as follows: unit weight 117-188 lbs/cu ft; natural moisture content 0-38%; porosity 0.15; Young's modulus 2 X 10(E+3)-6.8 X 10(E+6) psi; Poisson's ratio 0.03-0.50; dilation angle +5 to -5 degree; uniaxial compressive strength 70-37,000 psi; tensile strength 0-1540 psi; coefficient of linear thermal expansion 4 X 10(E+6)/F; horizontal permeability 10(E-8)-10(+3) ft/yr; and vertical permeability 1/2-/10 times horizontal permeability. It is apparent that a span of several orders of magnitude exists for certain key parameters and that virtually any rock-mass behavior could be predicted, if the

<22> CONT.
appropriate parameters were chosen. The primary conclusion to be drawn from this information is that a thorough investigation of the specific shale formation must be carried out before a rational decision to construct an underground repository in that particular material can be made. Attempts to determine the nature and variability of the rock mass must be carried out. Tests must also be performed on the shale from the depths of interest (1000 to 3000 feet) under loading conditions which simulate the actual conditions which are expected (such as high confining stresses and high temperatures). (Auth) (LKM)

Porosity; Young's Modulus; Poisson's Ratio; Compressive Strength; Tensile Strength; Thermal Expansion; Permeability

REVIEWS; SHALES; SEDIMENTARY ROCKS; PHYSICAL PROPERTIES; THERMAL PROPERTIES; REPOSITORY; SITE SELECTION; ROCKS; ILLITE; EXCAVATION; STORAGE, GEOLOGIC; ROCK MECHANICS

<23>
Dames and Moore, White Plains, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 7. Baseline Rock Properties - Basalt.

Y/OWI/TR-36/7; 148 pp.(1978, April)

This report contains an evaluation of the results of a literature survey to define the rock mass properties of a generic basalt, which could be considered as a geological medium for storing radioactive waste. The general formation and structure of basaltic rocks is described. This is followed by specific descriptions and rock property data for the Dresser Basalt, the Amchitka Island Basalt, the Nevada Test Site Basalt and the Columbia River Group Basalt. Intact properties of these basalts ranged as follows: unit weight 150-192 lbs/cu ft; Young's modulus 5.06 X 10(E+6)-13.5 X 10(E+6) psi; porosity 0.19-8.6%; Poisson's ratio 0.19-0.32; bulk modulus 8.76 X 10(E+6) psi (measured for the Amchitka Island Dense Basalt only); shear modulus 3.72 X 10(E+6) - 6.02 X 10(E+6) psi; uniaxial compressive strength 21,860-63,900 psi; tensile strength 1900-3500 psi; coefficient of linear thermal expansion 1.6 X 10(E-6)-3 X 10(E-6)/F; heat capacity 0.17-0.2 BTU/lb-F; thermal conductivity 0.58-0.78 BTU/hr-ft-F; permeability 1.0-8 X 10(E-3) ft/yr. Data from the literature on other specific and non-site specific areas are also included. Since very few field tests and measurements of large-scale field behavior have been made, a considerable amount of engineering judgement was necessary in arriving at these rock-mass properties. Given the present state-of-the-art and the available field information, these properties are considered to be reasonable but are not unique. Before a repository is designed, and built, a major program of laboratory and field testing will be necessary to determine the necessary rock mass properties for that particular site. (Auth) (LKM)

Young's Modulus; Poisson's Ratio; Porosity; Bulk Modulus; Shear Modulus; Compressive Strength; Tensile Strength; Thermal Expansion; Heat Capacity; Thermal Conductivity; Permeability

REVIEWS; BASALTS; IGNEOUS ROCKS; ROCKS; PHYSICAL PROPERTIES; THERMAL PROPERTIES; REPOSITORY; SITE SELECTION; STORAGE, GEOLOGIC; ROCK MECHANICS

<24>
Dames and Moore, White Plains, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 20. Thermomechanical Stress Analysis and Development of Thermal Loading Guidelines.

Y/OWI/TR-36/20; 230 pp.(1978, April)

A thermo-mechanical analysis of proposed preconceptual repositories in granite, shale and basalt has been conducted on three different scales: i) Very Near Field (canister scale), ii) Near Field (excavation scale) and iii) Far Field (regional scale). Three numerical methods were used to undertake the thermo-mechanical calculations: the finite element method for thermal stress analysis, the boundary element method for thermal and thermal stress analysis, and the semi-analytical method also for thermal and thermal stresses analysis. The properties and stratigraphic sections of the generic rock types have been compiled from available literature and used as the input data base. Because of the limited data on rock mass properties, assumptions where necessary on rock mass behavior have been incorporated into the analyses. Very Near Field analysis indicated that within 1/2 year of burial, fracturing of the rock adjacent to the canister will occur in all rock types for thermal loads as low as 1.2 kW/canister. From past experience, it was assumed that fracturing would extend about 6 in. into the rock mass, provided a steel liner was installed in the storage borehole. Near Field studies of room and pillar stability recommend acceptable thermal loads of 190, 120, and 190 kW/acre, for a 5-year retrievability period in granite, shale, and basalt, respectively, for 25-year retrievability, maximum thermal loads are 100, 55, and 100 kW/acre, respectively. Far Field analysis shows maximum ground surface uplifts less than 2 ft and surface temperature rises less than 1 deg. F at times of 500-7000 years after disposal. Calculated temperature rises at the repository level are 350 F for granite and shale and 300 F for basalt containing spent fuel; rises of 250 F for granite and 200 F for shale and basalt are calculated for HLW. Preliminary recommendations are made for further investigations. (Auth) (LKM)

REVIEWS; STRESS; PHYSICAL PROPERTIES; THERMAL PROPERTIES; ROCK MECHANICS; GRANITES; SHALES; BASALTS; IGNEOUS ROCKS; SEDIMENTARY ROCKS; ROCKS; REPOSITORY; SITE SELECTION; MODELS, MATHEMATICAL; MODELS; FRACTURES; TEMPERATURE; WASTES, HIGH-LEVEL; SPENT FUELS; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; WASTE STORAGE; WASTE DISPOSAL; STRUCTURAL STABILITY; THEORETICAL STUDIES

<25>
Dawson, P.R., and J.R. Tillerson, Sandia Laboratories, Computational Physics and Mechanics Division, Albuquerque, NM.

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CHEMICAL AND PHYSICAL ASPECTS

<25> CONT.
Nuclear Waste Canister Thermally Induced Motion.

SAND-78-0566; 31 pp. (1978, June)

A thermodynamically-coupled finite element model of viscous flow and heat transfer in salt due to heat-generating radioactive waste canisters has been developed. Temperature-dependent thermal conductivity was included in the analyses, with flow fields and temperature distributions coupled through temperature-dependent body forces, temperature-dependent constitutive equations, viscous dissipation, and changes in system geometry. Free expansion of the salt with temperature rises was assumed, leading to the largest density differences and therefore the greatest driving forces for upward salt flow. Separate thermoelastic computations were performed to evaluate the validity of this free expansion assumption. Creeping viscous flow calculations were done using the COUPLEFLO computer code, and the thermoelastic calculations with the SANDIA-BRINES program. An axisymmetric region 500 m in radius extending 750 m above and below a canister horizon was analyzed. In the transient creeping flow and heat transfer analyses, the heat source started at 313 K and diminished at a rate based on a 30 year half-life. Steady-state creeping flow and heat transfer were used to predict upper bounds on the magnitude of velocities. For the steady-state analyses, the velocity field and thermal distribution calculations assumed constant thermal output at 313 K. Results of coupled transient analyses using constant salt viscosity ($\mu = 0.5 \times 10^{15}$ Pa-sec) projected over 10 years, showed the canister sinking, then rising as its heat generates a convective cell. Maximum canister velocity during the 10 year period was 1.5 μm/sec; total displacement was 0.0001 m. Transient analyses with temperature-dependent salt viscosity have been carried out for times up to 150 years. Downward canister velocity was increased due to reduced salt viscosity after heating; after 35 years the velocity became positive, peaking at 0.1 μm/sec upward. After 125 years, convective cell velocity diminished, and the canister sank again. Total displacement over 150 years was 0.0003 m. Steady-state analyses for a number of canister densities, variable and constant salt viscosities, etc. yielded a maximum upward velocity of 26.5 μm/sec. Displacement of the surface at the repository centerline as predicted by the thermoelastic model was a maximum of 0.15 m after 30 years. (LKH)

See also Salt Motion Following Nuclear Waste Disposal, Dawson and Tillerson, January 1978.

MODELS; MODELS, MATHEMATICAL; STORAGE, GEOLOGIC; SALT DEPOSITS; WASTES, HIGH-LEVEL; HEAT TRANSFER; HEAT SOURCES; CREEP; THEORETICAL STUDIES; COMPUTER CODES; THERMAL PROPERTIES; VISCOSITY; CANISTERS

<26>
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An Evaluation of Kriging Techniques for High-Level Radioactive Waste Repository Site Characterization.

PNL-2903; 74 pp. (1979, January)

The purpose of this study is to evaluate kriging, a statistical technique that has potential use in site selection and safety assessment of high-level radwaste storage. Kriging is a statistical method for estimating functions that describe spatially-distributed phenomena such as groundwater elevation (potential) and depth to basalt. A potential geologic repository must be evaluated from very sparse data. Core samples are intrusive and costly; thus, the objective is to minimize the number of cores taken. The contribution of kriging to this objective is two-fold. First, kriging produces a contour model of the geologic formation of a potential site with an associated measure of uncertainty. This provides a more complete characterization than can be gained from other contouring methods. Secondly, given the existing data, kriging can be used to optimize the selection of additional sampling locations. This feature provides a rationale for sample-location selection and could cut data acquisition costs significantly. To evaluate kriging by experimenting with the techniques on representative data sets, the Hanford Wells data base was used. Water elevation tends to be a fairly smooth spatially-distributed phenomenon and since the Hanford Wells water elevation data set is fairly extensive, the September 1973 and January 1975 data were used to gain familiarity with the techniques. Kriging computations are performed using BLUEPACK, a computer package developed by the Paris School of Mines and installed at the DOE-Las Vegas computing facility. The basic criterion for evaluating kriging in this study is how realistically it contours the water potential and basalt surfaces. The kriged contours of the water potential surface based on only the well data are quite similar to the hand-drawn contours (standard error less than 5 ft for most of the area studied). The basalt surface is much less well-known than the water potential surface; consequently, the basalt kriging standard errors were generally an order of magnitude larger than those for the water elevation surface. This reflects the more complex spatial structure of the basalt deposits. The main conclusion of this study is that kriging can be a useful tool in developing realistic contours of the geologic structure of a potential repository site. Second kriging standard errors provide a quantification of the precision of the contours; this can be useful evidential information for developing a site selection rationale. Thirdly, insights gained from kriging analysis can contribute to the substantive knowledge of the geologic or hydrologic phenomenon under study. (Auth) (LKH)

THEORETICAL STUDIES; STATISTICS; SITE SELECTION; MAPS; WATER TABLE; STRATIGRAPHY; DEPTH; COMPUTER PROGRAMS; COMPUTER CODES; GEOLOGY; HYDROLOGY; METHODS; STORAGE, GEOLOGIC; REPOSITORY; WASTE STORAGE

<27>
Eliason, J.R., Battelle-Pacific Northwest Laboratories, Richland, WA.

Montmorillonite Exchange Equilibria with Strontium-Sodium-Cesium.

American Mineralogist 51:328-335. (1966, March-April)

<27> CONT.
The ion exchange isotherms and the free-energy changes of -2278, -331, and -4019 cal/mole for the cesium-sodium, strontium-sodium, and cesium-strontium systems respectively were determined for the Bayard, New Mexico montmorillonite. The cation exchange capacity for the Bayard montmorillonite was found to be 1.17 meq/g. The capacity was determined using sodium, strontium and cesium as the cation. The ion exchange isotherm and the free-energy change of -2035 cal/mole for the cesium-sodium system on the Chambers, Arizona montmorillonite were determined for comparison with the Bayard montmorillonite. The sodium-cesium system of the Bayard montmorillonite showed a surprising increase in the selectivity for cesium above 90% cesium loading. The selectivity for strontium was found to increase over the entire range of strontium loading in the sodium-strontium system on the Bayard montmorillonite. Over the range from 0 to 40% strontium loading the montmorillonite was sodium selective. Above 40% it became strontium selective, with a marked increase in strontium selectivity above 90% strontium loading. The effect of increased selectivity with increased loading of cesium and strontium on the sodium clay may be due to the difference in the c-axis spacing noted in the Bayard montmorillonite in the X-ray diffraction work. (JET)

Ion Exchange Capacity

Cs; Sr

MONTMORILLONITE; SORPTION; LABORATORY STUDIES; CATIONS; CHEMICAL PROPERTIES; ION EXCHANGE; ION EXCHANGE CAPACITY; SODIUM; DIFFRACTION, X-RAY

<28>
Erdal, B.R., R.D. Aguilar, R.P. Barkhurst, R.P. Daniels, C.J. Duffy, F.O. Lawrence, J. Naestas, F.C. Oliver, and K. Wolfsberg, Los Alamos Scientific Laboratory, Los Alamos, NM.

Sorption-Desorption Studies on Granite. 1. Initial Studies of Strontium, Technetium, Cesium, Barium, Cerium, Europium, Uranium, Plutonium, and Americium.

LA-7856-MS; 62 pp. (1979, February)

Distribution ratios were determined for sorption-desorption of radioactive tracers between the Climax Stock granite (quartz monzonite porphyry) obtained at the Nevada Test Site and a water prepared to be reasonably representative of the natural composition of water in equilibrium with the Climax Stock granite. The measurements were performed at 22 C and 70 C under atmospheric oxygen conditions. Elements given in order of increasing distribution coefficient at ambient temperature are: U(+6), Sr, Tc(+7), Ba, Ce(+3), Cs, Eu(+3), Pu, and Am. At 70 C the order is: Tc(+7), Sr, Ce(+3), Eu(+3), Ba, Cs, Pu, and Am. The effects of surface area and mineralogy on sorption were also investigated. The sorption ratio [r(d)=activity per g of solid/activity per ml of water] increases slowly with time presumably due to the slow interaction of the rock surface with the groundwater. Surface alterations presumably are also the reasons for the observation that it is more difficult to remove Cs, Ce, Eu, Pu, and Am from the rock than it was to sorb these elements (desorption ratio greater than sorption ratio). The sorption ratios for all elements studied, except Ce and Eu, scale with surface area or particle size and increase with temperature. This presumably indicates that Ce and Eu do not follow an ion-exchange or sorption mechanism. No appreciable sorption of Tc(+7) and U(+6) was observed. Permeability and porosity measurements were also made on consolidated cores. (Auth) (WAF)

Sorption ratios of all elements measured are given in graphic and tabular form.

Distribution Coefficient; pH

Sr 85; Tc 95m; Cs 137; Ba 133; Ce 141; Eu 152; U 237; Pu 237; Am 241

SORPTION; DESORPTION; GRANITES; ROCKS; TRACERS; TEMPERATURE; SURFACE PROPERTIES; MINERALS; PERMEABILITY; POROSITY; LABORATORY STUDIES; CATION EXCHANGE CAPACITY; PARTICLE SIZE; GROUND WATER; EQUATIONS; MINERALOGY; AGPILITES; STRONTIUM 85; TECHNETIUM 95M; CESIUM 137; BARIUM 133; CERIUM 141; EUROPIUM 152; DISTRIBUTION COEFFICIENT; pH; CORES; TIME FACTOR; SHALES; SEDIMENTARY ROCKS

<29>
Swirg, B.B., and J.V. Morrigan, University of California, Sanitary Engineering Research Laboratory, Berkeley, CA.

Ion Exchange Properties of a Representative Oil Sand.

Ground Disposal of Radioactive Wastes, R.J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA. (pp. 131-136), 168 pp. (1961, July)

Investigations were conducted on the Meyer sand, a "typical" oil sand, which was selected as suitable for radioactive waste disposal. The Pliocene Meyer sand was sampled at a depth of 8,500 feet in the Santa Fe Springs Field, California. The air permeability of the core samples was 600 millidarcies and the porosity, 38%. The specific gravity was 2.59, and the chemical composition was 30-35% quartz with the rest feldspars and micas. The exchange capacity as determined by the ammonium acetate method ranged from 1.9 - 2.7 meq/100 g. Three Meyer sand columns 19 mm in diameter were saturated with a solution of 0.1 N calcium nitrate, then saturated with a solution of 0.1 N calcium chloride labeled with Ca 45 and 1.7 X 10(P-8) N cesium chloride labeled with Cs 137. Following saturation, the Ca 137 and Ca 45 were eluted with a solution identical to the saturating solution. The breakthrough of the chloride ion was used to determine the column porosity, while the breakthrough of Ca 45 made possible the measurement of exchange capacity. The total exchange capacities (summation of Ca and Cs) were 2.40, 2.20, and 2.19 meq/100 g, for the 30.0, 119.8, and 236.0 cm sand columns. Porosity of the sand columns ranged from 31.46 to 33.3%. The mass-action constant in ml/g for the sand columns were 926 for the 30 cm column, 906 for the 119.8 cm column, and 531 for the 236 cm column. Tests were conducted to determine the influence of calcium concentration on Cs exchange on Meyer sand. The total exchange capacities varied from 3.0246 meq/l at 1.0 meq/ml calcium nitrate to 0.3166 meq/ml calcium nitrate. Kaolin was selected for

<29>

CHEMICAL AND PHYSICAL ASPECTS

<29> CONT.

farther study of Cs exchange. The exchange capacity was 0.0595 meq/g. At a [Cs]/Ca of 0.0073, cesium demonstrated approximately the same affinity for kaolin (K sub Cs-Ca = 54) as for Hanford sand (K sub Cs-Ca = 74). The influence of hydrogen ion on cesium exchange was investigated. K sub Cs-H was found to be highly dependent on pH, increasing sharply with increasing hydrogen concentrations. When the pH was 3.35, K sub Cs-H was 29.1 and when pH was 6.1 K sub Cs-H was 45.5. Indications are that in the case of kaolin, changes in hydrogen ion concentrations of over six orders of magnitude may be expected to reduce the capacity of the exchanger for cesium by only about one order of magnitude. On the other hand, at nearly neutral pH, hydrogen ions appear to have a far greater affinity for kaolin than cesium when K sub Cs-H is used as a criterion of sorption preference. (PTC) (NOT)

Porosity; Permeability; Grain Size Distribution; Ion Exchange Capacity; pH

Cs

BREAKTHROUGH DISTRIBUTION; CALCITE; CATION EXCHANGE CAPACITY; EQUILIBRIUM CONSTANT; LABORATORY STUDIES; MASS BALANCE; PERMEABILITY; POROSITY; pH; RADIONUCLIDE MIGRATION; KAOLIN; ROCK-FLUID INTERACTIONS; SAND; WASTE DISPOSAL

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Ewing, R.C., University of New Mexico, Department of Geology, Albuquerque, NM.

Metamict Mineral Alteration: An Implication for Radioactive Waste Disposal.

Science 192(4244):1336-1337, (1976, June 25)

A possible method for evaluating the long-term stability of silicate or borate glasses used for radioactive waste disposal is to examine metamict minerals for alteration effects. Although the exact mechanism for the transition from crystalline to metamict state is poorly understood, radiation damage from constituent U and Th nuclides must be critical to the process. Thus metamict minerals provide an ideal model for studying glasses that have suffered radiation damage over geolog. periods of time. The most detailed studies have been of complex Nb-Ta-Ti oxides [AB2O6-type, where A = REE, Fe(+2), Mn, Ca, Th, U, or Pb, and B = Nb, Ta, Ti, or Fe(+3)], in which alteration is of two types: 1) primary or hydrothermal, and 2) secondary or weathering. With primary alteration there is a consistent increase in Ca, generally a decrease in U and Th, and a decrease in REE; there is a slight decrease in B-site cations and an increase in structural and absorbed water. Secondary alteration produces a decrease in Ca, an increase of leaching of A-site cations, and a relative increase in B-site cations. Both types of alteration are pervasive and affect physical properties; in some cases a spherulitic recrystallization is initiated. Examination of specimens in the U.S. Museum of Natural History (yttrialite, USNM 85070 and USNM 3780; gadolinite, USNM 5102; and allanite, USNM 96970) suggests that the effects are equally pervasive in metamict silicates. Autoradiographs reveal clear evidence of U and Th leaching along the microfractures common in metamict minerals. Detailed microprobe analysis of an altered zone in metamict yttrialite from the type Ranch pegmatite in Texas shows primary alteration with an increase in CaO and H2O and a decrease in SiO2, TiO2, ThO2, and REE. Although preliminary, the data suggest that a glass phase is susceptible to alteration and thus may be an inappropriate medium for radioactive waste disposal. (Auth) (EFN)

GLASS; MINERALS; LEACHING; MODELS; LABORATORY STUDIES; STABILIZATION, CHEMICAL; RADIATION EFFECTS; URANIUM; THORIUM; OXIDES; HYDROTHERMAL ALTERATIONS; WEATHERING; SILICATES; WASTE DISPOSAL; WASTES, RADIOACTIVE, VITRIFIED; WASTE STORAGE

<31>

Fried, S., A.M. Friedman, and P. Weber, Argonne National Laboratory, Chemistry Division, Argonne, IL.

The Distribution of Plutonium in a Rock Containment Environment.

CONF-740286; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, W.W. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1973. American Chemical Society, Washington, DC, Ch. 9, (pp. 126-133), 166 pp. (1976)

Studies were made of the migration of solutions containing plutonium through cores of Niagara limestone and basalts. A small amount of Pu 239 Pu(NO3)4 tracer in neutral aqueous solution was placed on the surface of the disk of stone and allowed to dry at room temperature. Successive increments of H2O were then forced through the limestone disk, and the depth of penetration of the plutonium tracer was measured by the x-ray absorption technique. The results of these experiments yielded a migration coefficient, m=13 plus or minus 10 (13)(E-6)m/m) of water flow for the limestone and 67 plus or minus 9 (1)(E-6)m/m) for the basalts, where m is the average distance traveled by the plutonium atoms for every meter traveled by the H2O molecules. A second set of experiments consisted of measurements of the surface absorption coefficient of plutonium on the stones. In these experiments disks of stone were immersed in solutions of 0.00004M Pu 239 Pu(NO3)4. Small aliquots (0.05M) of the solutions were removed, dried on Ta platchets, and then placed in an internal alpha proportional counter. When the counting rate of the samples taken at 12-hr intervals had become constant, this was regarded as evidence of equilibrium. The adsorption constant, k, was defined as: k = (activity of Pu/ml of solution)/(activity of Pu/square cm of stone). The value of k for pure solutions of Pu(NO3)4 at 0.00004M was 0.10 plus or minus 0.02 for limestone and 0.07 plus or minus 0.02 for basalts. The k value is affected by other ions in the solution, such as NaCl, CaCl2, LaCl3, and ZrCl4. It was found that at least two chemical forms of plutonium were present in the neutral solutions and one of these, presumably a polymerized plutonium oxide, migrated ten times faster than the other form. (JNT)

PLUTONIUM; BASALTS; LIMESTONES; IGNEOUS ROCKS; SEDIMENTARY ROCKS; RADIONUCLIDE MIGRATION; ADSORPTION; SALTS; LABORATORY STUDIES; CHEMICAL PROPERTIES

<32>
Friedman, A.M., and S. Fried, Argonne National Laboratory, Argonne, IL.

Importance of the Chemical Species Formed in the Interaction.

"/OSI/SDB-77/18268; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 39-40), 92 pp.(1977, August 20)

Laboratory studies conducted on the migration of actinides and long-lived fission products through rock have shown that since negative ion binding sites are much less prevalent in geologic strata, any negative species will tend not to be strongly absorbed and will thus migrate rapidly. Results have indicated that the higher oxidation states of the actinides [Pu(+6), Np(+5), and U(+6)] are stable and at least somewhat anionic in groundwater. In addition, I-, IO3-, and TcO4- are, of course, anionic and all of these ions have characteristically low values of absorption on many minerals. In contrast, the lower oxidation states of the actinides and Tc are found to migrate at a much slower rate. Present research efforts are concentrated on determining the situational characteristics of the chemical states and on methods of identifying which states are present in very dilute solutions. Techniques which have been developed include electrophoresis and mixed tracer reactions. Laboratory measurements are being made of adsorption and desorption kinetics of actinides and long-lived fission products in representative geologic strata (basalts and tuffs), and on the relative reversibility of the process. Flow and movement of ionic species along fissures, colloid formation and transport, and precipitation and solution mechanisms are being examined. Experimental techniques involve use of both oxygen saturated and oxygen-free solutions. Some absorption coefficient values have been determined for Pu, Np, Tc, and I species. (ERG)

Only the abstract is given.

F Np; Tc; I

LABORATORY STUDIES; ROCKS; IGNEOUS ROCKS; BASALTS; TUFFS; ACTINIDES; FISSION PRODUCTS; RADIONUCLIDE MIGRATION; COLLOIDS; TRACERS; OXIDATION; ADSORPTION; ABSORPTION; SORPTION; DESORPTION; PLUTONIUM; TECHNETIUM; NEPTUNIUM; IODINE; DISTRIBUTION COEFFICIENT; GROUND WATER; CHEMICAL PROPERTIES; CHEMICAL ANALYSIS; IONS; IONIC PROCESSES; RADIONUCLIDE MIGRATION; WASTE-ROCK INTERACTIONS; METHODS

<33>
Gedeyne, M.P., R.E. Grim, and T.F. Zeiger, University of Illinois, Department of Geology, Urbana, IL.

Illite: A Model Based on the Sorption Behavior of Cesium

American Mineralogist 51:1684-1696, (1966, November-December)

Samples composed of the clay mineral illite and illite-mixed-layer material in the less than 2 micron fraction were used to investigate the sorption behavior of Cs. These samples, the Beavers Bend, Rock Island, Marblehead, and Pithian illites, and "Grundite", varied moderately in chemical composition and the amount of mixed-layering present as diagnosed by X-ray diffraction techniques. The total cesium sorption capacity of each sample was obtained by batching 1.0 g of sample with 100 ml of cesium nitrate solution ranging in concentration from 1 to 10(E-10)M. No competing cations were introduced in the solutions in order to better follow the clay-cesium reactions during the analyses. Results of the radiochemical tests of the sorption capability of the illites show that maximum percentages of cesium sorbed occurred within a relatively short period of time, usually within four hours. 75 to 90% of the cesium was sorbed by the clay in the lowest concentration range. The maximum sorption recorded in this range was for Pithian illite, followed closely by "Grundite", and Rock Island, Marblehead, and Beavers Bend illites. The amount of mixed-layering in these samples may be directly correlated to the cesium sorption capacity, as Pi[illegible] an illite and "Grundite" contain the greatest evidence of mixed-layering, and also have the highest sorption capacity. Diffraction analyses of these treated samples showed that the primary effects of cesium-sorption are a slight broadening and damping of the 001 intensities, with no appreciable changes in the basal spacings. One-dimensional structural projections of the illite samples before and after appropriate treatments with cesium-bearing solutions indicate that at the levels of cesium concentration in the contacting solutions, no structural changes have taken place which can be defined by X-ray diffraction analysis. This an interpretation may be made that with cesium sorption at concentration levels which apparently exhibit evidences of a high sorption capacity of the 10 angstrom clay for cesium, no structural incorporation can be conclusively established. (27)

Percent Adsorption

Cs

ILLITE; SORPTION; DIFFRACTION; X-RAY; LABORATORY STUDIES; PARTICLE SIZE; SOILS; CLAYS; CESIUM; LATTICE

<34>
Serx, F., Oak Ridge National Laboratory, Oak Ridge, TN.

Disposal of Radioactive Wastes in Salt Domes.

Health Physics 29:1-7, (1975, July)

Diapiric salt structures, which are caused by the plastic deformation of thick salt beds, have been proposed as possible locations for the disposal of radioactive wastes requiring long-term containment. There are, however, several important questions concerning tectonic stability, salt dissolution, and reliability of long-term containment that would have to be satisfactorily answered before this scheme of waste disposal could be considered feasible. Several concepts have been proposed for the emplacement of radioactive wastes in diapiric salt structures. The most promising of these are: (1) a mechanical mine, and (2) the mixing of granular waste and crushed salt in deep solution cavities. In comparison with a mechanical mine, a granular mixture in

<34>

CHEMICAL AND PHYSICAL ASPECTS

<34> CONT.

solution cavities would require the mining of larger volumes of salt and would be feasible only with wastes aged significantly more than 10 yr; on the other hand, it could offer greater reliability of long-term containment because of the increased isolation provided by the greater depth. (Auth)

THEORETICAL STUDIES; WASTES, HIGH-LEVEL; WASTE DISPOSAL; SALT DOMES; REVIEWS; DIAPIRS; TECTONICS; GROUND WATER; SEDIMENTS; THERMAL PROPERTIES; REPOSITORY; CAVITIES; SIZES; DEPTH; DRILLING; WASTES, LIQUID; CONTAINMENT; SITE SELECTION

<35>

Haug, H.O., Institut fur Heisse Chemie, Gesellschaft fur Kernforschung mbH, Karlsruhe, German Federal Republic.

Some Aspects and Long-Term Problems of High-Level and Actinide-Contaminated Spent Fuel Reprocessing Wastes from the Uranium-Plutonium and Thorium-Uranium Fuel Cycles.

IAEA-SM-207/51; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976, Vol. 2. International Atomic Energy Agency, Vienna, (pp. 233-242), 828 pp. (1976)

A comparison is made of the long-term radiotoxicity of different high-level and actinide-containing wastes from nuclear fuel reprocessing and of naturally occurring radioactive materials. The radiotoxicity of the radionuclide inventory in solid or solidified wastes fixed in a practicable insoluble matrix is compared with the radiotoxicity of the nuclide inventory of the same amount of uranium ore (0.2 wt% U), which consequently allowed the definition of a relative toxicity index. The comparison is extended to high-level waste disposal of single bore holes in deep geological formations. The results indicate that after about 1000 years of decay the radiotoxicity of the controlling actinides in the total volume of the high-level waste disposal layer will not be higher than the radiotoxicity of the same amount of a low grade uranium ore deposit containing 0.2 wt% U. The concentration limitations present by the single bore hole disposal geometry can also be met by similar disposal conditions for other actinide-containing wastes. (Auth)

ACTINIDES; WASTES, HIGH-LEVEL; FUEL REPROCESSING; TOXICITY; BOREHOLES; WASTE DISPOSAL; STORAGE, GEOLOGIC; HAZARD ANALYSIS; THEORETICAL STUDIES

<36>

Heard, H.C., and A. Duba, Lawrence Livermore Laboratory, Livermore, Cal.

Stress, Strain, and Temperature Induced Permeability Changes in Potential Repository Rocks.

T/CONF/SWB-77/10260; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 43-46), 92 pp. (1977, August 20)

Terminal storage of high-level radioactive wastes requires placement of the solidified materials in a stable environment devoid of circulating groundwater. This implies initially that the potential repository rock is of low permeability and that subsequent to excavation of the vault, this permeability will not increase excessively. It also implies that the thermal field resulting from the emplaced waste canisters will not lead to structural problems or large permeability increases with time. The goal of this work is to assess the permeability characteristics of coarse-grained igneous rocks as affected by pressure, deviatoric stress, and temperature. In order to predict the long-term behavior of these rocks, both virgin and fractured, permeability and all principal strains resulting from an imposed deviatoric stress under various simulated lithostatic pressures are being measured. In addition, compressional as well as shear velocities and electrical conductivity along these principal directions are being evaluated. These simultaneous measurements are being made initially at 25 C on a 15 cm diameter by 30 cm long sample in a pressure apparatus controlled by a mini-computer. The equipment is capable of generating a 2 kbar confining pressure at temperatures up to 300 C while producing an axial load on the core of 25-30 kbars. The entire vessel is heated. Correlation of these data with similar field observations should then allow simplified exploration for a suitable repository site as well as the prediction of the response of a mined cavity with both distance and time at this site. After emplacement of the waste canisters, the mechanical stability and hydrologic integrity of this mined repository will be directly influenced by the fracturing of the surrounding rock which results from local temperature differences and the thermal expansion of that rock. Temperatures (and, hence, these differences) in the vicinity of the repository are expected to be affected by the presence of pore fluids (single- or two-phase) in the rock, the heat capacity, and the thermal conductivity of this system. In turn, these are all dependent upon lithostatic pressure, pore pressure, and stress. Thermal expansion (and fracturing) will also be affected by the lithostatic (and effective) pressure, the deviatoric stress field, and the initial anisotropy of the rock. (Auth) (LKH)

Only the abstract is given.

PHYSICAL PROPERTIES; THERMAL PROPERTIES; WASTES, HIGH-LEVEL; WASTES, SOLID; PERMEABILITY; STRAIN; STRESS; FRACTURES; PRESSURE EFFECTS; LABORATORY STUDIES; FIELD STUDIES; SITE SELECTION; REPOSITORY; WASTE-ROCK INTERACTIONS

<37>

Mickey, C.E., Jr., Sandia Laboratories, Fluid and Thermal Sciences Department, Albuquerque, NM.

Steady Thermal Convection at Low Rayleigh Number from Concentrated Sources in Porous Media.

SAND-77-1529; 48 pp. (1977, December)

In this report, several mathematical models are proposed for the analysis of steady thermal convection from a concentrated heat source in a fluid-saturated porous medium, as part of an assessment of the sea bed as an effective barrier to release of disposed

<37> CONT.

radwastes. In general, radionuclide transport in this type of medium involves 4 basic processes: convection, diffusion-dispersion, sorption, and radioactive decay. In order to model this transport it is necessary to provide some estimate of the fluid velocity; this is the motivation for the analysis presented in this report. Approximate descriptions of the temperature and velocity distributions are presented for a variety of idealized situations involving localized heat sources embedded in a fluid-saturated porous medium. In all cases, it was assumed that the associated Rayleigh number was small enough so that the effect of convection on temperature distribution could be neglected. The idealized situations studied included a point source, vertical line source, and isothermal sphere embedded in an unbounded porous medium. Solutions were also obtained for the same sources when they were embedded in a porous medium a finite distance below a plane surface over which both pressure and temperature were constant. The latter situation is more representative of the conditions expected to prevail in the sea bed. In all cases, both temperature and velocity distributions were determined and representative isotherms and streamlines were plotted. Pressure distributions can be obtained by integration of the velocity components. Analytical estimates of the vertical transport of fluid particles by convection were obtained for both steady and decaying point sources. An approximate thermal power decay rate based on calculations reported by E.D. Klett (1977) was used in the analysis. The results presented can be readily extended to situations involving arrays of point sources, vertical cylinders, or spheres. A model for a horizontal line source of finite length could also be obtained. (Auth) (LEH)

THEORETICAL STUDIES; MODELS; MODELS, MATHEMATICAL; HEAT SOURCES; HEAT EXCHANGE; RADIONUCLIDE MIGRATION; HYDRAULIC VELOCITY; TEMPERATURE; THERMAL PROPERTIES; SEA DISPOSAL; WASTE DISPOSAL; WASTES, HIGH-LEVEL

<38>

Hinkebein, T.E., and P.F. Hlava, Sandia Laboratories, Albuquerque, NM.

Microstructural Interactions of Geologic Media with Waste Radionuclides.

SAND-78-0108; 28 pp. (1978, March)

A preliminary investigation of the microstructure of four geologic media was undertaken to examine the interaction of Cs, Sr, Gd and U with these media. Since Kd measurements often vary by more than an order of magnitude in rocks of the same strata separated by several feet, it was probable that minor rock components play an important role in radionuclide sorption. A qualitative analysis of the rock surfaces after equilibration with solutions of Cs(+1), Sr(+2), Gd(+3), and UO2(+2) revealed that clay minerals are responsible for all important sorption that takes place on the rocks studied (Magenta dolomite, Bell Canyon silt stone, Elgana shale and clay bearing halite). Thus it is concluded that a relatively small portion of these rocks is actively responsible for sorption characteristics. Gd was found on all samples, probably as Fe bearing chlorites. Cs was strongly held on the Elgana shale, probably by an illite. Although uranium was associated with an unidentified clay in the halite sample, uranium sorption probably occurred as UO2CO3 which would not significantly retard uranium movement at trace concentrations. Similarly Sr was retained only as SrSO4 in Magenta and this mechanism is not significant at trace levels. These analyses were performed by correlating electron microprobe elemental distribution photomicrographs with x-ray analyses of the four rocks. (Auth)

Fifty-seven elemental distribution photomicrographs of test items and individual minerals are given.

pH; Eh

CESIUM; STRONTIUM; GADOLINIUM; URANIUM; OXIDES; GEOLOGIC STRUCTURES; SORPTION; DOLOMITE; SILTSTONES; SHALES; IRON; CHLORITES; ILLITE; CLAYS; IONS; CHEMICAL ANALYSIS; SULFATES; MINERALS; SAMPLES; LABORATORY STUDIES

<39>

Hinkebein, T.E., and P.F. Hlava, Sandia Laboratories, Albuquerque, NM.

Interactions of Simulated Waste Radionuclides and Rocks.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 819-822), 563 pp. (1979)

Several sedimentary rocks were used in a study of radionuclide solution and rock interactions. Chip samples from the Magenta dolomite, Bell Canyon sandstone, Elgana shale, and clay-bearing Salado salt were equilibrated with solutions of Cs, Sr, Gd, and UO2. Distribution coefficients were measured and the chips were examined with an electron microprobe. Apparently as a result of the Fe-bearing chlorite Gd was sorbed on all samples, while only the Elgana shale retained Cs. This was undoubtedly due to the illite. A thick rind of UO2CO3 was formed on the Magenta dolomite and in the salt sample U was associated with pockets of fine-grained, clay-rich material. Strontium reacted with the CaSO4 to SrSO4 in the Magenta dolomite. Hence, it is reasonable to conclude that clay minerals in the rocks are responsible for the important sorption characteristics of the rocks. (NDY)

SORPTION; DOLOMITE; SHALES; SANDSTONES; SALTS; RADIONUCLIDE MIGRATION; WASTE-ROCK INTERACTIONS; CLAYS; MINERALS; LABORATORY STUDIES

<40>

Hlava, P.F., and W.F. Chambers, Sandia Laboratories, Albuquerque, NM.

Interaction of a Glass-Based Nuclear Waste Simulant and Natural Rock.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al.,

<80> CONT.
Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 265-270), 563 pp. (1979)

An experiment was conducted to determine the interaction between a glass containing simulated high-level radioactive waste and natural rock, to test a rock-melt concept of waste disposal in which wastes are ultimately encapsulated and immobilized in a solidified silicate mass at depth. The experimental procedure was insertion of a simulant glass containing stable isotopes into a hole bored into a 0.9 X 0.9 x 1.8 m block of microgabbro. A platinum-sheathed resistance heater melted the glass from day 21 to day 46 through a temperature range of 680 to 1350 C. Cooling began on the 48th day, and the glass resolidified on day 50. The block was sectioned and its elemental composition was determined by X-ray fluorescence. Color zonation was observed in the glass, indicating a possible convection flow boundary, so electron probe analysis was done for 17 elements, including all of those contained in the glass simulant and several important ones from the rock. The results showed that a convection cell formed in the molten region which kept the central part of the cell well mixed and homogeneous. There white band of pyroxene crystals forming a boundary between the stagnant zones and the convection cell. Four types of crystalline phases grew from the melt: 2 oxides, a borate, and a silicate. Spinel crystals were dispersed throughout the melt, and various other crystallites were observed. (Auth) (DCW)

The paper provides a schematic drawing of the zonation observed in the cooled melt, a table of elements analyzed by and standards used for the electron microprobe, and 2 photomicrographs of crystallites.

Temperature

WASTE DISPOSAL; WASTE STORAGE; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTE PROCESSING; WASTE-ROCK INTERACTIONS; STORAGE, GEOLOGIC; GLASS; ENCAPSULATION; IMMOBILIZATION; SOLIDIFICATION; SILICATES; THERMAL PROPERTIES; TEMPERATURE; LABORATORY STUDIES; METHODS; GABBROS; IGNEOUS ROCKS; CONVECTION; STABILIZATION, CHEMICAL; PYREX; ISOTOPES; RADIOACTIVE DECAY; DECAY HEAT

<81>
Jenks, G.H., Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Effects of Temperature, Temperature Gradients, Stress, and Irradiation on Migration of Brine Inclusions in a Salt Repository.

ORNL-5526; 67 pp. (1979, July)

This report reviews and analyzes available experimental and theoretical information on brine migration in bedded salt, including the effects of temperature, thermal gradient, stress, irradiation, and pressure. The theoretical and experimental (with KCl) results of Anthony and Cline were used to correlate and explain the available data for rates of brine migration at temperatures up to 250 C in natural crystals of bedded salt from Lyons and Hutchinson, Kansas. It was concluded that the following empirical equation for V/G sub s (migration velocity of brine inclusion per unit temperature gradient in the salt) represents the maximum value for this parameter which would result from thermal gradients in the bedded salt crystals of a repository: log V/G sub s = 0.00656T - 3.6036, where T is the temperature of the salt (C), and V/G sub s has the units (cm/yr/C). The most probable effects of stressing crystals of bedded salts on the migration properties of brine inclusions are a small fractional increase in the solubility of the salt within the liquid and a concomitant and equal fractional increase in the rate of thermal gradient-induced migration of the brine. Application of high pressure could reduce the value of the kinetic potential from that existing in the absence of the pressure, but this would not affect the maximum rates predicted by the equation. Results obtained in Project Salt Vault suggested that radiation energy stored within crystals had little, if any, influence on the rate of brine flow into emplacement cavities in salt. No direct information regarding the effect of stored energy on the solubility of salt is available; thus it is recommended that experiments be undertaken to provide such information. The greatest uncertainty relative to the prediction of rates of brine migration into a waste emplacement cavity is associated with questions concerning the effects of the grain boundaries on brine migration through the deposit. It is likely that grain boundary trapping will tend to retard brine migration under the conditions expected to prevail with probable repository designs. Brine inflow estimates for 10-yr old HLW emplaced at 150 kW/acre in salt containing 0.5 vol% brine inclusions indicated inflow rates starting at 0.7 liter/yr and totaling 12 liters at 30 yr after emplacement. Estimates for 10-yr old PWR SNFP emplaced at 60 kW/acre indicated a constant inflow of 0.035 liter/hr for the first 35 yr after emplacement. (Auth) (LKH)

Flow Rate

REVIEWS; THEORETICAL STUDIES; EQUATIONS; BRINES; SALT DEPOSITS; FLOW RATE; HYDRAULIC VELOCITY; THERMAL PROPERTIES; PRESSURE EFFECTS; STRESS; RADIATION EFFECTS; SOLUBILITY; REPOSITORY; STORAGE, GEOLOGIC; WASTE STORAGE; WASTES, HIGH-LEVEL

<82>
Jenks, G.H., and C.D. Bopp, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Storage and Release of Radiation Energy in Salt in Radioactive Waste Repositories.

ORNL-5058; 102 pp. (1977, October)

Experimental studies were conducted on the amounts of gamma-ray energy that are stored in salt under a variety of exposure conditions. Also investigated were thermal annealing characteristics of the stored energy in salt, the chemical reactions that radiation defects in salt undergo upon aqueous dissolution of the salt, and the retention of the radiation defects within the salt crystals. Results of the study showed that appreciable amounts of gamma-radiation energy can be stored under certain exposure conditions. It was also shown that thermally activated annealing takes place at elevated temperatures and that the rates of this annealing at temperatures above 150 C are such that negligible amounts of energy will

<42> CONT.

be stored in salt in a repository where the salt is a temperature above 150 C. No thermally activated annealing was observed in rock salt at temperatures below about 150 C. Some radiation-induced annealing might occur. The results of measurements of energy stored at irradiation temperatures between 30 and 150 C, together with results of theoretical considerations, showed that the maximum stored energy that would be formed in salt in a repository with no annealing whatever would be about 50 cal/g. No means was conceived by which the stored energy could be released abruptly, nor was any significant hazard believed to be possible from the release if it should occur abruptly by some unforeseen mechanism. In practice, salt temperatures below about 150 C will occur only with isolated or semi-isolated canisters in a repository, or with most canisters, after very long times during which the gamma-dose rate will have dropped off markedly. Available evidence from thermal annealing observations and from calorimetric and dissolution measurements indicated that there were no significant losses of either chlorine or sodium during or after irradiation. One effect of the stored energy which must be accounted for in a safety analysis is the generation of H2, which takes place upon aqueous dissolution of radiation-damaged salt, about 0.1 cu cm of H2 is generated per calorie of stored energy. However, no problems were recognized arising from this effect which cannot be counteracted by appropriate design and operation of the repository. (Auth) (RAP)

Sample materials used were bedded salt from the Carey mine at Lyons, Kansas, and from the Los Medanos area, New Mexico. Hershaw salt samples served as reference material.

pH

STORAGE, GEOLOGIC; SALT DEPOSITS; REPOSITORY; SALT DOMES; RADIATION, GAMMA; FRACTURES; THERMAL PROPERTIES; HAZARD ANALYSIS; CONTAINERS; DISSOLUTION; CHLORINE; SODIUM; WASTES, HIGH-LEVEL; DOSE RATE; RADIATION DOSE; HYDROGEN; LABORATORY STUDIES; RADIATION EFFECTS; EQUATIONS; IRRADIATION; CHEMICAL PROPERTIES; WASTE STORAGE; WASTES, RADIOACTIVE

<43>

Kazanjian, A.R., Rockwell International, Atomics International Division, Product Integrity and Surveillance Group, Rocky Flats Plant, Golden, CO.

Radiolytic Gas Generation in Plutonium Contaminated Waste Materials.

RFP-2469; 25 pp. (1976, October 29)

Many plutonium contaminated waste materials decompose into gaseous products because of exposure to alpha radiation. The gases generated (usually hydrogen) over long storage periods may create hazardous conditions. To determine the extent of such hazards, gas generation yields were measured by contacting some common Rocky Flats Plant waste materials with plutonium and monitoring the enclosed atmospheres for extensive periods of time. The materials were Plexiglass, polyvinyl chloride, glove-box gloves, machining oil, carbon tetrachloride, chlorothene VG solvent, Kimwipes (dry and wet), polyethylene, Dowex-1 resin, and surgeon's gloves. Gas generation rates were obtained in two series of tests: in one, materials were contacted with Pu 239 for 870 days, and in the other, materials were contacted with Pu 238 for 267 days. The gas analyses were made by mass spectrometry and the results obtained were the total gas generation, the hydrogen generation, the oxygen consumption rate, and the gas composition over the entire storage period. The gas yields, in units of standard cu cm per day per Ci of plutonium, varied from 16 for Texaco Regal-A machining oil to 0.71 for Dowex-1 ion exchange resin. The predominant gas in most of the materials was hydrogen. The oxygen consumption rates varied from 0.0008 to 0.070 mM/day per gm of plutonium oxide 239 and from 0.0018 to 0.0051 mM/day per mgm Pu 238. Estimation of the hazards created in storage drums by the measured gas rates was not attempted because of its complexity. The results obtained indicate that although significant over-pressurization is possible, the probability is low. The probability depends on the material and many other factors. Appreciable amounts of hydrogen are usually produced, but will not be accompanied by oxygen in a drum for most of the storage period. (Auth) (LKH)

Extensive tabular data are provided.

RADIOLYSIS; GASES; HYDROGEN; PLUTONIUM 239; WASTES, RADIOACTIVE; PLUTONIUM 238; SPECTROMETRY; POLYVINYL CHLORIDE; POLYETHYLENE; GLOVES; OIL; SOLVENTS; RESINS; WASTES, SOLID; WASTES, LIQUID; LABORATORY STUDIES; WASTE DISPOSAL; CONTAMINATION

<44>

Kelley, J.A., and R.H. Wallace, Savannah River Laboratory, Aiken, SC.

Procedure for Determining Leachabilities of Radioactive Waste Forms.

Nuclear Technology 30:47-51. (1976, July)

A new and improved procedure was developed for determining leachabilities of proposed radioactive waste forms. Finely divided glass samples are leached in flowing deionized water; leached ions are continuously sorbed on ion exchange resins to control ionic quality. Sorbed ions are eluted from the resin columns for atomic absorption analyses. Leachabilities measured by this procedure are lower and more consistent than those made in stagnant water without continuous ionic control. (Auth)

LABORATORY STUDIES; WASTES, RADIOACTIVE; LEACHING; GLASS; WATER; ION EXCHANGE; RESINS; IONS; SORPTION; SOLIDIFICATION; WASTE STORAGE; PARTICLE SIZE; pH

<45>

Klechkovsky, V.N., L.N. Sokolova, and G.N. Tselishcheva.

The Sorption of Microquantities of Strontium and Cesium in Soils.

A/CONF.15/P-2310; Peaceful Uses of Atomic Energy, Proceedings of the 2nd United Nations International Conference, Geneva, Switzerland, September 1-13, 1958. United Nations, Geneva,

<45>

CHEMICAL AND PHYSICAL ASPECTS

<45> CONT.
Switzerland, Vol. 18, (pp. 885-893).(1958)

Laboratory studies were conducted to determine the ability of soils to absorb strontium and cesium, and to compare the behaviors of Sr 90 with Ca 45 and Cs 137 with K 42 in soil. Three types of soil were used: the turf-podzolic soil of the Forest Experimental Station of the Timiryazev Agricultural Academy (Moscow), the black soil of the Suny Experimental Station (Ukrainian Republic), and the red soil from the Batumi area (Georgian Republic). Along with the radioactive isotopes of strontium and cesium used without carriers, the radioactive isotopes Ca 45 and K 42 were also used in the experiments. These isotopes were added in some experiments as a tag to active salt solutions to investigate a comparison of the sorption curves received after various periods of interaction between the soil and the active solution. The distribution coefficients for the three types of soils for strontium are as follows: Turf-podzolic = 565, Black soil = 1182, and Red soil = 628. The sorption of microquantities of strontium by the soils from salt solutions is markedly reduced, but with an increase in the concentration, the change in the value of the distribution coefficient is not strictly proportional to the salt concentration. From the CaCl2 solutions, the sorption of the Sr 90 by podzolic soils and Black earth is somewhat lower than the sorption from solutions of KCl. In the experiment with red soil, the sorption of Sr 90 from CaCl2 solution was somewhat stronger. With the cesium, the distribution coefficients of the three types of soils were 1094 for Turf-podzolic, 3017 for Black soil, and 1192 for the Red soil. The preliminary substitution of all the exchange cations for potassium had an unusual effect on the sorption of microquantities of cesium by soils. The sorption of cesium by red soil and turf podzolic soils increased considerably when all the exchange cations in the sorbing complex were substituted for potassium. In the experiment with black soil, the preliminary substitution of all the exchange cations by potassium scarcely changed the sorption of cesium, whereas its sorption by red soil went up 7-10 times as a result of such substitution. At the same time, it was noted that the sorption of cesium by soils without their preliminary saturation by potassium or calcium was many times greater if it took place from a solution of calcium chloride than from a solution of potassium chloride. (JST)

There is a considerable amount of tabular data on the distribution coefficients of the three soils studied in this report.

Distribution Coefficient; pH

Sr 90; Ca 45; Cs 137; K 42

ABSORPTION; CATION CONCENTRATION; CALCIUM COMPOUNDS; CHLORINE; COMPLEXES; CONTAMINATION; DISTRIBUTION COEFFICIENT; FOOD CHAINS; IMMOBILIZATION; ION EXCHANGE; ISOTOPES; LABORATORY STUDIES; POTASSIUM; POTASSIUM COMPOUNDS; POLLUTION, SOIL; SAMPLING; SOILS; UPTAKE

<46>
Kosiewicz, S., B. Barraclough, and A. Zerwekh, Los Alamos Scientific Laboratory, Los Alamos, NM.

Degradation of RAD Wastes.

LA-7582-PR; Studies of Transuranic Waste Storage under Conditions Expected in the Waste Isolation Pilot Project (WIPP), July 1-September 30, 1978, (pp. 3-11), 16 pp.(1979, January)

This is the fourth quarterly report describing progress in an experimental program to determine the effects of radiolytic, thermal, and catalyzed pyrolytic degradation of various forms of existing or proposed transuranic (TRU) wastes. Thermal degradation of paper and composite (35% cellulosics, 23% polyethylene, 12% polyvinyl chloride, 15% neoprene, and 15% Hypalon) at 70 and 100 degrees C is occurring. Some radiolytic gas evolution appears to be occurring at a dose of 4 X 10(E+5) aCi/g. The rates of sludge dewatering at 25, 40, 70, and 100 degrees C have been determined. Several high-pressure (4.5 X 10(E-8) kPa, or 150 atm) experiments are in progress. In addition, an assessment of Los Alamos Scientific Laboratory (LASL) retrievably stored TRU waste for the calendar years 1971 to 1977 is in progress. The total volume of TRU waste stored at LASL during these years is 1589 cu m or about 3.5% of the national Department of Energy total. (Auth)

Waste Volume

LABORATORY STUDIES; WASTES, TRANSURANIC; WIPP; THERMAL PROPERTIES; RADIOLYSIS; GASES; SLUDGES; PRESSURE EFFECTS; WASTE VOLUME; RADIATION EFFECTS; DEGRADATION; PLASTICS; LABORATORY STUDIES

<47>
Kosiewicz, S.T., B.L. Barraclough, and A. Zerwekh, Los Alamos Scientific Laboratory, Los Alamos, NM.

Studies of Transuranic Waste Storage Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), October 1-December 15, 1978.

LA-7609-PR; 15 pp.(1979, March)

This report describes progress in an experimental program to determine the effects of radiolytic, thermal, and catalyzed pyrolytic degradation of various forms of existing or proposed transuranic contaminated wastes. Preliminary data appear to indicate that G(gas) is dose-rate independent. The rate of evolution of radiolytically generated gases decreases with increasing exposure time. More gas is evolved by radiolysis at 70 C than at 20 C. A summary of other work on alpha radiolysis of organics is provided. The rates of cement paste dewatering have been determined at 40 and 100 C. Characterization of (1) waste types, (2) waste containers, and (3) radionuclide occurrences are presented in detail in the Los Alamos Scientific Laboratory TRU waste inventory assessment. At the end of 1977, 27 waste types comprised the total inventory of retrievably stored, solid, TRU contaminated waste. The total accumulated volume amounted to 1590 cu m, 20% of which is combustible. The total TRU waste inventory is stored in 2813 containers. Five different types of containers are in use: 210-l drums, 110-l drums, CHP, TRP crates, and cement casks. The total curie content of the TRU waste inventory is estimated to be close to 8.0 x

<87> CONT.
10(E+8) Ci with Pu 238, Pu 239, and Am 241 dominating the inventory's curie content. (RAP)

WASTES, TRANSURANIC; WASTE STORAGE; RADIOCHEMISTRY; WIPP; DEGRADATION; GASES; DOSE RATE; TEMPERATURE; ORGANIC COMPOUNDS; CEMENTS; CONTAINERS; DRUMS; CASKS; RADIOLYSIS; LABORATORY STUDIES; ALPHA PARTICLES; WASTE VOLUME; URANIUM 233; URANIUM 235; URANIUM 238; NEPTUNIUM 237; PLUTONIUM 238; PLUTONIUM 239; AMERICIUM 241; AMERICIUM 242; CURIUM 244; CALIFORNIUM 249; ACTINIUM 227; THORIUM 232

<88>
Lin, W., Lawrence Livermore Laboratory, Livermore, CA.

Measuring the Permeability of Eleana Argillite from Area 17, Nevada Test Site, Using the Transient Method.

UCRL-52604; 11 pp.(1978, December 11)

Using the transient method, permeability of high-quartz Eleana argillite from the Nevada Test Site was determined as a function of effective pressure. By comparing calculated and observed pressure decay in the upstream reservoir, the permeability of intact and fractured specimens have been determined at effective pressures from 1.0 to 20.0 MPa. Over this range, Eleana argillite has a low permeability [(10(E-16)-10(E-19) sq cm)] when intact and a higher permeability [10(E-12)-10(E-17) sq cm] with one induced through-going fracture. Although the fracture does increase permeability significantly, the increase is not prohibitively great at high pressure. For a reasonably deep repository (500-2000 m), the presence of fractures would not necessarily exclude storing nuclear wastes safely. From the viewpoint of water transport at room temperature, high-quartz Eleana argillite seems a viable alternative to either Westerly or Barre granite as a location for a radioactive waste repository. Within the argillite, permeability seems to decrease with increasing depth. It was not possible to measure permeability on a low-quartz specimen; the effect of low quartz content on permeability of the Eleana argillite must be further studied. The effects of temperature, time, and anisotropy on permeability must also be investigated. (Auth)(LKH)

Permeability

SHALES; SEDIMENTARY ROCKS; PERMEABILITY; MEASUREMENTS; FRACTURES; SITE SELECTION; STORAGE, GEOLOGIC; GROUND WATER; QUARTZ; COMPUTER CODES; MODELS; MODELS, MATHEMATICAL; LABORATORY STUDIES; THEORETICAL STUDIES; REPOSITORY; PRESSURE EFFECTS

<89>
Lin, W., Lawrence Livermore Laboratory, Livermore, CA.

Compressible Fluid Flow Through Rocks of Variable Permeability.

UCRL-52308; 16 pp.(1977, July 25)

The effectiveness of coarse-grained igneous rocks as shelters for burying radioactive wastes can be assessed by determining the rock permeabilities at their in situ pressures and stresses. In this report, analytical and numerical methods are used to solve differential equations of one-dimensional fluid flow through rocks with permeabilities from 10(E+8) to 1 nD. In these calculations, upstream and downstream reservoir volumes of 5, 50, and 500 cu cm were used. The computer code TRUMP was used to obtain numerical solutions of the variation in pressure for both the upstream and downstream reservoirs over time. The optimal size combinations of the two reservoirs were determined for measurements of permeability, stress, strain, acoustic velocity, and electrical conductivity on low-porosity, coarse-grained igneous rocks. Optimal combinations for measuring permeability up to 10(E+3) nD are (upstream reservoir volume, downstream reservoir volume): (50, 50), (500, 50), (50, 500), and (500, 500). The optimal reservoir combinations for permeabilities less than or equal to 10(E+3) are (50, 50), (50, 500), and (500, 500). (Auth)(LKH)

PERMEABILITY; THEORETICAL STUDIES; PRESSURE EFFECTS; STRESS; ROCKS; IGNEOUS ROCKS; EQUATIONS; FLUID MECHANICS; COMPUTER CODES; STRAIN; METHODS; WASTES, RADIOACTIVE; WASTE DISPOSAL; NUMERICAL METHODS; POROSITY

<90>
Lincoln, R.C., D.W. Larson, and C.E. Sisson, Sandia Laboratories, Albuquerque, NM.

Estimates of Relative Areas for the Disposal in Bedded Salt of LWR Wastes from Alternative Fuel Cycles.

SAND-77-1016; 247 pp.(1978, January)

The relative nine-level disposal areas required for 5 types of LWR radioactive wastes disposed into a hypothetical horizontally-bedded salt formation have been estimated. Waste types considered were unreprocessed spent reactor fuel [SF(UO2)], solidified waste from reprocessing of U-oxide fuel [HLW(UO2)], Pu from reprocessing spent reactor fuel doped with 1.5% of the accompanying waste from reprocessing U-oxide fuel(SPK Pu), waste from reprocessing of mixed U/Pu-oxide fuel in the third recycle[HLW(RCY)], and unreprocessed spent fuel after 3 recycles of mixed U/Pu-oxide fuels[SF(ROX)]. The model used in the calculations assumes no compaction or subsidence of strata over the repository, no thermal effects due to aquifers, one-dimensional vertical heat flow only, waste emplacement layers represented by flat cylindrical discs, earth's surface as an isothermal boundary, and the 4000 m deep bottom of the repository as an adiabatic boundary; potential repository operational constraints were not considered, and no stress calculations were taken into account. The model site is composed of 5 layers; from top to bottom, layer 1 is a mix of sandstone, limestone, shale, and surface sediments; 2 is half salt interbedded with anhydrite, shale, and/or dolomite; 3 is salt (disposal horizon); 4 is identical to 2; and 5 is a mix of sandstone, limestone, and shale. Five cases are modeled: The Reference Case, Deeper Case (same thermal properties and salt volume, but disposal horizon is 3/2 deeper), Low-K Case (thermal conductivity of layers 1

<50>

CHEMICAL AND PHYSICAL ASPECTS

<50> CONT.
and 5 are 2/3 of Reference Case), Less-Salt Case (layers 2, 3, and 4 are 2/3 thickness of Reference Case, but disposal depth and thermal properties are the same), and Variable-Properties Case (thermal conductivity of layers decreases with increasing temperature). Estimated waste disposal area ratios determined from calculated values for the maximum thermal energy (BTU) content of the formations are as follows [HLW(WO2) = 1 in all cases]: Reference Case - 8.9 for SF(WO2), 8.3 for SPK Pu, 2.6 for HLW(HOX), and 11 for SF(HOX); Deeper Case - 6.5 for SF(WO2), 5.7 for SPK Pu, 3.3 for HLW(HOX), and 14 for SF(HOX); Low-K Case - 5.6 for SF(WO2), 4.8 for SPK Pu, 2.9 for HLW(HOX), and 12 for SF(HOX); Less-Salt Case - 5 for SF(WO2), 4.3 for SPK Pu, 2.6 for HLW(HOX), and 11 for SF(HOX); Variable-Properties Case - 8.5 for SF(WO2), 3.9 for SPK Pu, 2.4 for HLW(HOX), and 10 for SF(HOX). (LKH)

STORAGE, GEOLOGIC; GEOLOGIC STRATA; WASTES, HIGH-LEVEL; THERMAL PROPERTIES; THERMAL CONDUCTIVITY; HEAT TRANSFER; HEAT EXCHANGE; SALT DEPOSITS; SEDIMENTARY ROCKS; SHALES; SANDSTONES; LIMESTONES; ANHYDRITE; MODELS; THEORETICAL STUDIES; WASTE STORAGE; WASTE MANAGEMENT

<51>
Lomenick, T.F., Oak Ridge National Laboratory, Oak Ridge, TN.

Laboratory Pillar Model Experiments.

ORNL-4555 (Draft); ORNL-CF-76-7-82; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 12, (pp. 12.1-12.92), 822 pp. (1977, April)

Scale models of salt pillars and their surrounding rooms were fabricated and tested under conditions that might be expected in actual operations, to study the deformational behavior of rock salt in mined openings. Laboratory tests were conducted at temperatures ranging from 22.5 degrees to 200 degrees for axial loads of from 2000-10,000 psi for each temperature. The deformation of the pillars increased markedly with increasing loads and temperature, but should decrease with time. Thin shale beds within the salt deposits caused the accelerated rates when they occurred at roof and floor interfaces. Other rock salts from various locations were used in order to make general comparisons of deformational behavior. United States deposits from which samples were taken are: Lyons and Hutchinson (Kansas); Retsof (New York); Detroit (Michigan); Grand Saline (Texas); and Cote Blanche (Louisiana). The Asse salt was the European example. At 22.5 C deformation of the Lyons, Hutchinson, Michigan, Cote Blanche and Asse were similar. The Grand Saline salt showed a higher deformation rate than the others while the Retsof salt showed an initially slower rate. The reasons for the differences are not fully understood. Mechanism of deformation are discussed. (CAJ) (NDV)

REPOSITORY; SALT DEPOSITS; MINES; DESIGN; LABORATORY STUDIES; THERMAL PROPERTIES; EVALUATION; ROCK MECHANICS; DEFORMATION; STRESS; EVALUATION; SHALES; MODELS; STRUCTURAL STABILITY

<52>
Martin, R.T., Massachusetts Institute of Technology, School of Engineering, Cambridge, MA.

Feasibility of Sealing Boreholes with Compacted Natural Earthen Material, Volumes 1 and 2.

ORNL/SUB-3960/2; MIT Research Report 75-28, Volumes 1 and 2; 206 pp. (1975, June)

This report describes laboratory investigations of borehole sealing using compacted shale or shale component materials in a shale substrate. Materials have been examined for permeability, strength, shrinkage, and swell pressure to estimate feasibility of both the materials and possible construction techniques. Three requirements for feasibility are: (1) permeability less than 0.1 microdarcy; (2) adhesion between the plug and shale formation sufficient to resist a 300-ft water head; and (3) durability for 200,000 years. Laboratory compacted plugs can be made that meet the permeability and shear strength requirements for Dewey Lake Shale, Minnewah Shale, and montmorillonite. To achieve the permeability required necessitated very high compactive effort (equivalent to 4000 psi static pressure) for a montmorillonite plug in Dewey Lake Shale. The maximum constant volume swell pressure measured on laboratory compacted samples was 680 psi. Allowing 1% strain reduced the swell pressure 50%. Compacted samples allowed to shrink by air drying to constant weight gave a maximum volumetric shrinkage of 3.8%. Permeability tests on shrinkage samples after soaking while confined to the original compacted volume gave no change in permeability due to shrinkage and reswell. Based on the laboratory tests it was concluded that crushed shale core material, shale outcrop material, and montmorillonite are, within reason, feasible materials for sealing boreholes. Additional research is needed to optimize any particular material with any particular compaction technique. Compaction methods that appear to be technically feasible are: static, impact, roller, and air gun. From the laboratory testing done, permeability appears to be the critical parameter. Therefore permeability measurement during field testing will be important not only for proving satisfactory field test construction, but also for proving quality assurance through correlation with data obtained from an instrument package on the compactor. A major point of vital importance toward achieving an optimum final product is the very strong interdependence of the material, the construction method, and the instrument package necessary for quality control of the construction and of the final product. (Auth) (LKH)

Volume 2 consists of appendices detailing the work described in Volume 1.

BOREHOLES; SEALING; SHALES; SEDIMENTARY ROCKS; STORAGE, GEOLOGIC; LABORATORY STUDIES; MINERALOGY; REPOSITORY; PHYSICAL PROPERTIES; MONTMORILLONITE; CLAYS; SEDIMENTS; EVALUATION; COMPACTION; PERMEABILITY; STRENGTH, MECHANICAL; QUALITY ASSURANCE

<53>
Hartiner-Doerr, L.F., and C.H. Radick, Lawrence Berkeley Laboratory, Berkeley, CA; University of California, Petroleum Engineering Laboratory, Berkeley, CA.

Thermal Properties of Cable Mountain Basalt Cores Hanford Nuclear Reservation.

LBL-7038; 8 pp. (1978, March)

Thermal conductivities and diffusivities of a group of basalt cores from the Cable mountain area were evaluated. Thermal conductivities were measured, using a steady-state comparator apparatus, at a fixed stress and at two temperature levels. Specific heats were calculated as a function of temperature from oxide analyses. Bulk densities were measured at room conditions; densities at elevated temperatures were estimated from available thermal expansion data. From the above data, thermal diffusivities were calculated. (Auth) (NAP)

Thermal Conductivity; Density

LABORATORY STUDIES; THERMAL PROPERTIES; THERMAL CONDUCTIVITY; DENSITY; SITE SELECTION; BASALTS; THERMAL DIFFUSIVITY; SPECIFIC HEAT

<54>
McCarthy, G.J., S. Komarneni, D.E. Scheetz, and W.B. White, The Pennsylvania State University, Materials Research Laboratory, and Department of Geosciences, University Park, PA.

Hydrothermal Reactivity of Simulated Nuclear Waste Forms and Water-Catalyzed Waste-Rock Interactions.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 329-340), 563 pp. (1979)

A key consideration in the long term safe disposal of nuclear wastes is their stability in the repository environment. If the assemblage of waste phases is not in thermodynamic equilibrium with the mineral assemblage of the host rocks, chemical reactions which are greatly enhanced by the presence of water or brine may occur. Closed-system experiments with three waste forms (all nonradioactive simulations)--spent fuel, a glass, and a crystalline ceramic, and three rock type--basalt, shale, and salt, were done in sealed gold capsules contained in autoclaves or cold-seal pressure vessels, with the pressure at 300 bars, the temperature range 200 to 300 C, and in the presence of excess water so that all phase assemblages included a liquid phase whose composition was determined by the relative solubility of different components. The experiments illustrated three types of hydrothermal waste-rock interactions: reactivity of waste forms in the presence of aqueous solutions (distilled water or WIPP-6a bittern brine); reactivity among waste, rock, and water; and potentially important radionuclide fixation reactions. The solution that resulted from the experiments with spent fuel and deionized water contained Cs, Rb, Mo, and U, but no Sr, Ba, La, or Nd. The WIPP-6a brine dissolved Cs, Rb, large amounts of Sr, Ba, La, and Nd, and 40 times more U than the deionized water. Less Mo was in solution because it reacted with the Ca in the brine to form powellite ($CaMoO_4$). The results of the experiments with borosilicate glass and deionized water or brine showed that while glass was very reactive in both deionized water and brine, the former converted the glass to a crystalline mineral-like product, while in the latter, major amounts of all the important radionuclide model elements were extracted and remained in solution. In the experiments with the crystalline ceramic, deionized water had very little effect but brine dissolved large percentages of key elements in the ceramic, including Cs and Sr. However, after as much as two months of treatment, several phases, including monazite, were still unaltered, which supports the suggestion that monazite is a potentially ideal host for nuclear waste actinides. Also, the ability of bittern brines to extract from waste forms so much of the heat producing Sr and Cs isotopes, along with substantial U, needs to be given serious consideration in repository selection and engineering design. When basalt was added to the spent fuel-deionized water experiments, most of the Cs was removed from solution. Hence, the availability of oxygen fugacity buffering such as that found in basalts and many shales appears to be an important attribute of a geological repository. Mechanisms for the fixation of Cs were determined to be formation of pollucite and powellite. More attention to optimized synthetic minerals for particular repositories is needed. (Auth) (DCW)

Four tables give percents of elements in solution after hydrothermal treatments; two figures show X-ray diffractograms of borosilicate glass before and after hydrothermal treatment, and the products of hydrothermal interactions between cesium and labradorite.

WASTES, RADIOACTIVE; REPOSITORY; MINERALS; HOST ROCKS; CHEMICAL PROPERTIES; WASTE-ROCK INTERACTIONS; SPENT FUELS; GLASS; CERAMICS; BASALTS; SHALES; SALT DEPOSITS; LABORATORY STUDIES; GEOCHEMISTRY; FISSION PRODUCTS; RADIONUCLIDES; CHEMICAL COMPOSITION; ACTINIDES; FIXATION; BRINES; MONAZITE; CESIUM; STRONTIUM; URANIUM; NEODYMIUM; LANTHANUM; DECAY HEAT; EXPERIMENTS; BARIUM; RUBIDIUM; MOLYBDENUM; CALCIUM

<55>
McClain, W.C., and A.M. Stanfield, Oak Ridge National Laboratory, Oak Ridge, TN.

Analysis of Combined Effects.

ORNL-4555 (Draft); ORNL-CF-70-7-82; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 13, (13.1-13.59), 822 pp. (1971, April)

A quantitative analysis of the deformational behavior of the entire experimental area under the combined influence of stress, stress history, heat, thermal gradients and time is presented. Numerical values are used for physical parameters in contrast to the qualitative interpretations presented in chapter 11. A computer program was written which combined various influences over a broad area, and repeated computer runs were made and the various effects observed. Based on the accuracy of the computer models and the reasoning that went into them, it seems that the objective of the measurement and analysis has been achieved: the combined

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CHEMICAL AND PHYSICAL ASPECTS

<55> CONT.
physical effects on the deformation of salt are understood well enough that a rational design of actual waste disposal facilities can be undertaken. (CAJ)

Equations and graphs are included.

STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; MINES; DISPOSAL SITE; DESIGN; DEFORMATION; EVALUATION; STRESS; ROCK MECHANICS; MINES; SITE EVALUATION; RECOMMENDATIONS; THERMAL PROPERTIES; COMPUTER PROGRAMS; THEORETICAL STUDIES; EQUATIONS; THERMAL PROPERTIES

<56>
McEvoy, M.B., Lawrence Berkeley Laboratory, Real-Time Systems Group, Berkeley, CA.

Data Acquisition, Handling, and Display for the Heater Experiments at Stripa.

LBL-7062; SAC-18; 71 pp. (1979, February)

In June 1978, a joint Swedish/American research team began acquiring data from the Stripa mine in Sweden, 360 m below the surface. Electrical heaters were installed and are now used to assess the suitability of granite rock as a repository for radioactive waste material. Extensive instrumentation measures temperature, stress, and displacement effects caused by these heaters, which are intended to simulate nuclear waste canisters in size and power output. A data acquisition system is used to acquire and digitize data, convert measured values to engineering units, store data for random access, provide on-site graphical data displays, and transfer data to Lawrence Berkeley Laboratory for further analysis. Duplicate computer peripherals and backup data loggers were used to reduce data losses caused by failure during the planning 1.5 years of operation. This report describes the data acquisition system, its design considerations, capabilities, and operational use. The techniques employed to detect and analyze any anomalous experimental results are also described. Many design considerations were the result of instrument capabilities and characteristics, and the desired accuracies imposed by our research requirements. The system also required additional support such as adequate power supply, protective enclosures, and fire protection. Relevant environmental considerations are described in the Appendix. (Auth) (NAF)

COMPUTER PROGRAMS; FIELD STUDIES; REPOSITORY; WASTES, HIGH-LEVEL; DESIGN; INSTRUMENTS; HEAT SOURCES; CONTAINERS; STRESS; FRACTURES; GRANITES; THERMAL PROPERTIES; ROCK MECHANICS; MODELS

<57>
McKinstry, W.A., Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Thermal Effects in Shales: Measurements and Modeling.

I/OSI/SUB-77/18268; CONF-770789; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977. (pp. 53-54), 92 pp. (1977, August 20)

This research concerns thermal and physical measurements and theoretical modeling relevant to the storage of radioactive wastes in shale. Reference thermal conductivity measurements are made at atmospheric pressure in a commercial apparatus; equipment for permeability measurements has been developed. Thermal properties of shales are being determined as a function of temperature and pressure. In one test a 15 mm disk of sample is measured by a steady state technique using a reference material (initially single crystal quartz) to measure heat flow within the system. The sample is sandwiched between two disks of reference material and the thermal conductivity of the sample at steady state is calculated. A second test determines the effect of temperature on a larger sample of water-saturated shale (or siltstone). A cylindrical sample 25 cm diameter x 20 cm, is heated electrically at the center while contained in a pressure vessel that maintains a fixed water pressure around it. The temperature is monitored by a microcomputer system connected to 16 thermocouples to record variation in temperature distribution with time. The microcomputer system as currently devised can also control the energy delivered to the heating element. The vapor pulse should thus be measured without heating the outer wall of the pressure vessel. If, as hypothesized, the heat drives the water away from the internal heater the temperature distribution should reflect a dramatic temperature pulse passing through the sample. By using thermal conductivity data from the small sample experiments, permeability measurements, and the results of a finite difference computer model of the heat transfer, the behavior of the sample should be properly interpreted. (Auth) (LKH)

Only the abstract is given.

SHALES; MODELS; MODELS, MATHEMATICAL; THERMAL CONDUCTIVITY; PERMEABILITY; THERMAL PROPERTIES; PRESSURE EFFECTS; SEDIMENTARY ROCKS, WATER; GASES; HEAT TRANSFER; LABORATORY STUDIES; THEORETICAL STUDIES; WASTE-ROCK INTERACTIONS

<58>
Menzel, R.G.

Accumulation and Movement of Fission Products in Soils and Plants.

TID-7668; Use of Inorganic Exchange Materials for Radioactive Waste Treatment, D.K. Jamison, et al. (Eds.), Proceedings of a Working Meeting, Washington, DC, August 13-14, 1962. U.S. Atomic Energy Commission, Division of Technical Information, Washington, DC. (1963, January)

The effect of temperature on the fixation of Sr 85 in soils was investigated. Soils used in the study are Davidson, Dunbar, Maury, and Huntley. The first two soils probably fix Sr by adsorption and diffusion into amorphous material, while the Maury may fix Sr by precipition or recrystallization of phosphates. The Huntley, a calcareous soil, may fix Sr by precipitation or recrystallization of carbonates. For the Dunbar, a sandy loam, more Sr is fixed with increasing time and amount of Sr added. However, the percentage of Sr fixed decreases with the amount added. When the Dunbar was

<58> CONT.
incubated at 25 degrees C the amount of Sr sorbed increased from 30 to 80% for each threefold increase in time. At higher temperatures more rapid fixation occurred during the first 8 hours, with smaller increments thereafter. Leaching with hot acids indicates the sorption is largely a reversible reaction in a short time period. The rate-limiting step of the sorption reaction must involve the breaking of some chemical bonds on the surface of the soil particles, which is indicated by the reversibility of the reaction. More studies using the other three soils must be done before comparisons can be made with the Dunbar. (NDV)

SORPTION; ACIDS; CALCIUM CARBONATE; DECONTAMINATION; DISPOSAL SITE; FIXATION; LABORATORY STUDIES; STRONTIUM; LEACHING; LOAM; WASTE DISPOSAL; TEMPERATURE; WASTES, RADIOACTIVE; RECRYSTALLIZATION; PRECIPITATION, CHEMICAL; SOILS; WASTE-ROCK INTERACTIONS

<59>
Moore, J.G., E.T. Morgan, E.H. McDaniel, R.B. Greene, and G.A. West, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Cement Technology for Plugging Boreholes in Radioactive Waste Repository Sites: Progress Report for the Period October 1, 1977, to September 30, 1978.

ORNL-5528; 76 pp. (1979, August)

This program was initiated to develop and test cementitious mixtures suitable for plugging boreholes drilled during construction of a permanent radioactive waste repository. Experiments are in progress using 38 batches of cements from 13 different locales, representing 11 types of cement, and 17 batches of fly ash from 10 suppliers. Wet mix studies have revealed that addition of 10 wt% salt to a standard mortar will double the initial and final set times. Set time of the saltcrete can be increased or decreased by the addition of a lignite fly ash or a bituminous fly ash, respectively. Procedures were established for vacuum drying cementitious specimens to remove unbound water without causing damage. Drying is required for porosity measurements, as well as certain thermal conductivity and permeability measurements. Three methods were used to determine the initial porosities of selected specimens: saturated and dry densities, helium pycnometry, and mercury intrusion. Further studies must be made to determine which of these procedures is most advantageous. The compressive strengths of cement-fly ash concretes were found to vary with curing time and to depend on the composition of the fly ash. The strengths of saltcretes decreased with increasing salt concentration but increased with the addition of fly ash; the effect of the fly ash composition was apparently negligible. Thermal conductivities were found to increase with density and with sand and/or salt concentration. Measurements made over a range of 30 to 300 C showed that thermal conductivities of saltcretes decreased with increasing temperature but remained almost constant in neat cement pastes. Conductivity values ranged from 0.4 W/mK for a neat cement paste to 1.8 W/mK for a saltcrete. Gas and liquid permeability measurements are under way using a Hassler cell procedure. Results for a number of saltcretes under the same drying conditions showed that apparent liquid permeability decreased with increasing salt content; it varied from 6 X 10(E-7) darcy for 0 wt% salt to 1.25 X 10(E-9) darcy for a saltcrete containing 63.2 wt% salt. Initial liquid permeability measurements on concretes gave values about 2 orders of magnitude less than those calculated from gas measurements. Shrinkage measurements after an initial 24-hr curing period showed that, in general, shrinkage of neat pastes and saltcretes varied approximately linearly with time, while cement-fly ash solids showed a higher rate in the first few weeks followed by a lower rate for a longer time period. The reproducibility of these measurements was very poor; additional studies will be required to improve the reliability of the data and to assess the significance of the shrinkage results. (Auth) (LKG)

Thermal Conductivity; Permeability; Moisture Content; Density; Porosity; Compressive Strength

BOREHOLES; SEALING; CEMENTS; REPOSITORY; STORAGE, GEOLOGIC; FLY ASH; BITUMENS; SALTS; LIGNITES; BITUMENS; PERMEABILITY; THERMAL CONDUCTIVITY; POROSITY; CONCRETES; STRENGTH, COMPRESSIVE; SANDS; DENSITY; LABORATORY STUDIES

<60>
Moore, J.G., E. Morgan, and G.C. Rogers, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Radioactive Waste Fixation in FUETAP (Formed Under Elevated Temperature and Pressure) Concretes - Experimental Program and Initial Results.

ORNL/TM-6573; 21 pp. (1979, March)

This report describes studies now in progress to determine the technological feasibility and acceptability of concretes formed under elevated temperature and pressure (FUETAP) as hosts for radioactive wastes. The radioactive waste categories most amenable to this process and any technological and/or economic advantages over existing concepts for the fixation of high specific-activity wastes will be established. Initial studies show that reasonably hard, durable solids containing 10-15 wt% simulated radioactive sludge can be made by subjecting cementitious mixtures to 250 degrees C and 600 psi for 24 hr. Such solids exhibit extremely low plutonium leach rates and cause rapid recombination of hydrogen and oxygen at room temperature. Cements studied were Marquette Type 1, Atlas Type 3, and Universal Atlas HAC-Lumnite, as well as Kingston Fly Ash. Typical ranges of properties for the initial FUETAP concretes were: compressive strength 9.0-10.9 MPa; porosity (at 15,000 psi) 31.0-51.0%; density 1.40-1.56 g/cc cc; thermal conductivity (at 103 C) 0.43-0.53 W/mK; and water loss after 24 hr at 250 C, 12.1-62.66%. There was little difference in strengths of samples made with Type 1 or Type 3 cement, but specimens containing Lumnite, a high-alumina cement, had 30-40% less strength. Strength may be increased by increasing the fly ash/cement ratio or by increasing the time allowed to attain the desired curing temperature and pressure. In a specimen containing Type 1 cement and sodium nitrate, increasing the fly ash/cement ratio from 0.24 to 1.0 increased compressive

<60>

CHEMICAL AND PHYSICAL ASPECTS

<60> CONT.

strength from 14.2 to 18.7 MPa. Using the same materials but allowing 7 hr instead of 1.5 hr for the specimen to reach 250 C and 600 psi increased the compressive strength to 27.6 MPa. Studies are underway to determine the effect of these methods on the porosity, thermal conductivity, and water loss; future studies will relate these characteristics to the different preparation methods and compositions. It was observed that the presence of nitrate in the original concrete mixes significantly increased the amount of water loss at 250 C; because the presence of water could have a bearing on the amount of radiolysis in the final products, additional studies will be made to identify the mechanism behind this effect. Immersion of a specimen of Type I cement containing nitrate in simulated Sandia Type B brine for 8 days at 217 C and 300 psi caused no apparent change in the specimen, other than a 7.4% weight gain. Analysis of the brine showed that the major component lost from the specimen was nitrate (less than or equal to 85.8%); of the simulated sludge in the specimen, maximum loss was of Cr (less than or equal to 0.9%), followed by Ni, Al, and Fe. Additional studies are in progress in this area. (Auth) (LEN)

Thermal Conductivity; Porosity; Density; Compressive Strength

LABORATORY STUDIES; CEMENTS; CONCRETES; THERMAL PROPERTIES; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; SLUDGES; LEACHING; PLUTONIUM; STRENGTH, COMPRESSIVE; POROSITY; DENSITY; THERMAL CONDUCTIVITY; NITRATES; BRINES; CHROMIUM; IMMOBILIZATION

<61>

Morgan, H.T., Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Thermal Conductivity of Rock Salt from Louisiana Salt Domes.

ORNL/TM-6809; 18 pp. (1979, June)

The thermal conductivities of dome salt samples from the Avery Island and Jefferson Island Mines in southern Louisiana were determined as part of an evaluation of these formations for use as possible waste repositories. The thermal conductivities ranged from 4 W/m K at 70 C to 2 W/m K at 300 C. The average thermal conductivity of the dome salt at 100 C was about 25% less than that of a single-crystal salt at 100 C. The thermal conductivity was found to be related to the density and fragility of the salt. The thermal conductivities were measured using a comparative technique in which the sample was sandwiched between standard references of Pyroceram 9606. The precision of the method is within plus or minus 10% and the overall error is within plus or minus 20%. Considerable attention was given to specimen preparation. Finished surfaces were parallel within 0.005 inch and flat to within 0.0005 inch. Chromel-Alumel thermocouples were installed in grooves on interface surfaces to measure temperatures. (Auth)

Thermal Conductivity

THERMAL PROPERTIES; SALT DOMES; REPOSITORY; SAMPLES; DENSITY; POROSITY; EQUATIONS; MEASUREMENTS; METHODS; MOISTURE; THERMAL CONDUCTIVITY; [illegible] STUDIES

<62>

Morgan, H.T., J.G. Moore, R.E. Barney, G.C. Rogers, C. Williams, and E. Newman, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

The Disposal of Iodine 129.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 453-459), 563 pp. (1979)

Factors influencing the final disposal of I 129 are discussed. Iodine may be fixed in cement as barium iodate for permanent disposal. While permanent isolation cannot be guaranteed, a combination of isolation and dilution will prevent harmful effects to the populace. The leached fraction was linear with the square root of time, thus indicating a diffusion mechanism. Diffusion coefficients were reduced by a factor of 10 when 10% fly ash was added to the cement mixture. Annual production of I 129 is about 1/2 metric ton, and the maximum body burdens are 3 uCi in the thyroid and 200 uCi in the body. The normal content of I is 10 to 15 ug in the thyroid and 50 ug in the body. Therefore to reach the maximum permissible irradiation dose to the body an overdose would be required. The iodine content in sea water is 165 times the maximum permissible concentration. Two alternatives are available for permanent disposition of I 129: burial in a deep, dry geological repository, and disposal in the ocean. (NSW)

IODINE; CONTAINMENT; LEACHING; EQUATIONS; IMMOBILIZATION; DIFFUSION; MAXIMUM PERMISSIBLE CONCENTRATION; CEMENTS; DOSES; REPOSITORY; SEA DISPOSAL; BURIAL

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Morgan, H.T., and G.A. West, Oak Ridge National Laboratory, Chemical Technology Division, Nuclear Fuel and Waste Program, Oak Ridge, TN.

Thermal Conductivity of the Rocks in the Bureau of Mines Standard Rock Suite.

ORNL/TM-7052; 58 pp. (1980, January)

Thermal conductivities of eight rocks from the Bureau of Mines Standard Rock Suite were measured in air over the temperature range 373 to 533 K (100 to 260 C). The thermal conductivities of these rocks were measured to furnish standards for future comparisons with host rock from prospective nuclear waste repository sites. The rocks and their average thermal conductivities at 373 K before thermal cycling were (1) Sioux quartzite, 5.5 W/m deg K; (2) St. Cloud gray granodiorite, 2.7 W/m deg K; (3) Dresser basalt, 3.0 W/m deg K; (4) Barre granite, 2.5 W/m deg K; (5) Bolston limestone, 2.5 W/m deg K; (6) Westerly granite, 2.3 W/m deg K; (7) Berea sandstone, 2.2 W/m deg K; and (8) Salem limestone, 1.7 W/m deg K. The thermal conductivity at a given temperature decreased by as much as 9% after a specimen had been heated to the maximum temperature (533 K), but additional heating cycles had no further

<63> CONT.

effect. This decrease was smallest in the igneous rocks and largest in the sedimentary types. Variations due to orientation were within the precision of measurements (+ or - 5%). In most cases the thermal conductivities were linear with the reciprocal of the temperature and were within 10% of published data obtained by other methods. Measurements were made by a cut-bar comparison method in which the sample was sandwiched between two reference or metering bars made of Pyroceram 9606 glass-ceramic. The apparatus consisted of a Dynatech Model TCPCM-820 comparative thermal conductivity analyzer controlled by a Hewlett Packard Model 3052A data acquisition system. The four heater controllers in the Dynatech instrument were interfaced to the data acquisition system via a Hewlett Packard Model 69408 multiprogrammer. The computer control increased the instrument efficiency by more than a factor of 3 because overnight and weekend operation could be continued without surveillance and because computations were automatic. A program was written to increment and cycle the temperature in steps between predetermined initial and maximum values. At each step the thermal conductivity was measured after steady-state conditions were established. The rocks furnished by the Bureau of Mines were quarried in large and fairly homogeneous lots for use by researchers at various laboratories. To investigate any anisotropy, cores were taken from each rock cube perpendicular to each of the cube faces. Samples 2 in. in diameter and about 0.75 in. thick were prepared from the cores and were dried in a vacuum oven for at least one month prior to taking measurements. (Auth)

An appendix provides tables summarizing all sample temperatures, temperature differentials across sample and reference, measured thermal conductivities, and differences between measured and calculated thermal conductivities.

Thermal Conductivity

THERMAL CONDUCTIVITY; THERMAL PROPERTIES; QUARTZITES; GRANODIORITES; BASALTS; GRANITES; LIMESTONES; SANDSTONES; IGNEOUS ROCKS; SEDIMENTARY ROCKS; METHODS; EQUIPMENT; INSTRUMENTS; SAMPLES; MEASUREMENTS; LABORATORY STUDIES

<68>

Beamer, C.R., U.S. Geological Survey.

Ion Exchange Research by the United States Geological Survey.

Ground Disposal of Radioactive Wastes, W.J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, Ca, August 25-27, 1959. University of California, Berkeley, Ca, (pp. 117-119), 168 pp. (1961, July)

Cation ion exchange studies were conducted using clays as the exchange media. Vermiculite was used to scavenge Cs 137 and it was found that different samples varied greatly in the amount of time to exchange a specified amount of Cs 137. There was always a rapid initial exchange followed by a slower exchange. Kinetics studies indicate this is probably due to two different exchange sites. The initial exchange follows the course of a second-order reaction and the latter and slower exchange that of a first-order reaction. A study of the influence of pH on cesium take-up by vermiculite showed that at pH's above 3 cesium was completely removed from a 0.04 N cesium nitrate solution. A mixture of crandallite [$(CaAl_3(PO_4)_2(OH)_5)$], quartz, kaolinite, and illinite had the capacity to remove 0.3 to 0.6 mg of strontium per gram of ore from concentrations of 50 to 100 ppm strontium. Fundamental studies on clays to determine the different kinds of exchange sites were done by titration and ammonium acetate methods. Illite was shown to have only one exchange site while montmorillonite and vermiculite had two. The first site of montmorillonite has a capacity that is 62% of the total whereas the first site of vermiculite has a capacity of only 20% of the total. Further studies are underway. (NSW)

pH. Ion Exchange Capacity

Cs 137; Sr 90

BREAKTHROUGH DISTRIBUTION; CATION EXCHANGE CAPACITY; CLAYS; DECONTAMINATION; ION EXCHANGE; MONTMORILLONITE; VERMICULITE; ILLITE; CESIUM; STRONTIUM; WASTE TREATMENT; pH; SOLIDIFICATION; ENCAPSULATION; LABORATORY STUDIES

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Morris, A.E., Los Alamos Scientific Laboratory, Los Alamos, NM.

Oklo -- Natural Fission Reactor Program, Progress Report for July 1-September 30, 1979.

LA-8189-PR; ONWI/Sub/79/E511-01100-40; 6 pp. (1980, January)

The goal of this program is the determination of rates of reactor product migration and the mechanisms of transport in geologic media that include natural fission reactors or rich uranium ore bodies. This quarter nearly 200 samples were collected at the Oklo mine in Gabon for shipment to the U.S. to continue studies of lead, ruthenium, and technetium migration. Work is also underway to measure U and Pb concentrations and Pb isotopic abundances in samples selected from 402 samples which were collected at the Key Lake, Canada, uranium deposit. The purpose of this work is to measure the rate of radiogenic lead transport in a geologic regime distinctly different from that of Oklo. To date, U concentrations have been measured in 21 samples; Pb concentrations and isotopic abundances have been measured in 10 of these 21. Data indicate that samples from the Athabasca Sandstone and a pyrite-rich rock from the contact between the sandstone and crystalline basement rock are enriched in radiogenic lead relative to uranium. A pyrite-rich sample of basement rock underlying the uranium ore is relatively deficient in radiogenic lead. Additional work is needed before statements can be made about likely transport paths for this radiogenic lead. A technique was developed this quarter to eliminate the interference of organic materials during mass spectrometric analyses of Ru in Oklo samples with high asphaltic contents. In this technique, Ru is distilled from an ore solution, then ferric iron is added in sufficient quantity to permit its good recovery as the hydroxide. Sodium hydroxide is added to precipitate $Fe(OH)_3$, which carries the Ru. The solid phase is separated, dissolved, and redistilled, resulting in Ru yields as high

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CHEMICAL AND PHYSICAL ASPECTS

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as 98% -- free of organic interferences. A proposal has been drafted for a study of naturally occuring radionuclide migration at rich U ore bodies in Australia to be performed jointly by the U.S. Department of Energy and the Australian Atomic Energy Commission. (LEH)

REACTORS, NATURAL; ORE DEPOSITS; URANIUM; LEAD; POTASSIUM; TECHNETIUM; GEOCHEMISTRY; RADIONUCLIDE MIGRATION; ISOTOPES; METHODS; CHEMICAL ANALYSIS; LABORATORY STUDIES; SANDSTONES; SEDIMENTARY ROCKS; PYRITE; IGNEOUS ROCKS

<66>
Olsen, J.H., and R.T. Martin, Massachusetts Institute of Technology, Department of Civil Engineering, Constructed Facilities Division, Cambridge, MA.

Feasibility of Sealing Boreholes with Compacted Natural Earthen Materials, Volume 3.

ORNL/SUB-3960/2; MIT Research Report 76-26; 76 pp. (1976, June)

This report is a continuation of the investigation of sealing boreholes using compacted natural earthen materials against water intrusion for up to 200,000 years. Previous work showed that mechanical compaction in the laboratory can produce permeabilities less than 0.1 microdarcy (uD) in the following earth materials: (1) Dewey Lake Shale core material pneumatically impact compacted; (2) a mechanical mixture containing 20% montmorillonite and 80% Dewey Lake core material compacted by gravity impact; (3) a mixture of 1/3 montmorillonite and 2/3 Dewey Lake core material statically compacted with 9000 psi; and (4) montmorillonite statically compacted to give a dry density of at least 69 lbs/cu ft. Chemical stability of the montmorillonite phase is assured for any borehole temperature less than 75 C. The only possible change is conversion to illite, which requires potassium ions. Data from Gulf Coast sediments indicate that after 10 million years at 80 C, 40% expandable layers are retained. In view of the fact that montmorillonite consistently gave the lowest permeability of any material tested, it is concluded that it is the best possible material with which to make the seal section of a borehole plug. The compactive effort required to achieve low permeability is very modest static compaction (less than 800 psi). A system was built to measure changes in lateral stress during coring and subsequent plugging of a hole cored into shale. The system was designed so that lateral swell pressure and permeability could also be measured. Lateral swell pressure of intact Dewey Lake shale was 880 psi and 260 psi for the high and low clay content samples respectively. The coring operation gave a measured decrease in lateral stress at 1.3 hole diameters of 10 to 100 psi. Compaction at 8,000 psi and extrusion of the compacted plug from the high clay content shale sample gave a measurable degradation in lateral stress at 1.3 hole diameters. Similar tests at 1,000 psi on the high clay content shale and even at 6,000 psi compactive effort on the low clay content shale showed no degradation of lateral stress. The horizontal to vertical stress ratio, K, was measured over the applied vertical stress range of 0-800 psi. At the estimated in situ overburden stress the measured K was 1.9 and 0.85 for the high and low clay content samples respectively. Based on the laboratory results available, sandwich construction is recommended as the technique which permits optimization of materials and of quality control. Materials recommended for constructing the sandwich plug are montmorillonite clay for the seal section and quartz sand to provide mechanical stability to the seal section. The montmorillonite layer would be sandwiched between quartz sand layers. Compaction would be by static pressure on all layers. Sandwich construction using montmorillonite and quartz sand will provide the best possible borehole seal that can be achieved by mechanical means. Finally, the properties of the finished product will be more than adequate to keep water from the repository for 200,000 years. (Auth) (LEH)

Density

BOREHOLES; SEALING; REPOSITORY; STORAGE, GEOLOGIC; MINERALOGY; PHYSICAL PROPERTIES; MONTMORILLONITE; CLAYS; SEALS; SEDIMENTS; SEDIMENTARY ROCKS; EVALUATION; LABORATORY STUDIES; COMPACTION; PERMEABILITY; DENSITY; CHEMICAL PROPERTIES; QUARTZ; SANDS

<67>
Parimoos, E.G., RE/SPEC, Inc., Rapid City, SD.

Thermoelastic/Plastic Analysis of Waste-Container Sleeve: 1. Initial Estimates of Loading on the Sleeve.

ORNL/SUB-3706/6; 14 pp. (1976, May 9)

Waste container sleeve loading estimates have been obtained from an axially symmetric, thermoelastic/plastic, finite-element analysis of the SALT-6/1 model over a simulated 10 year heating period. Local failures of salt adjacent to the sleeve and in the vicinity of the excavated room are not observed; the model remains elastic throughout the excavation and heating periods. A reduced strength analysis indicates that a salt safety factor of approximately 2 is present. In this analysis, the computer code simulates a sequence of operations that includes: (1) excavation of typical repository room and drillhole in the room floor; (2) emplacement of the sleeve and waste container; and (3) subsequent heating over a ten year period. Material properties of the salt and waste container/sleeve arrangement assumed for this study were as follows: elastic modulus 0.46 X 10(E+6) psi for salt, 8.0 X 10(E+6) psi for the waste container/sleeve; yield strength 8500 psi for salt, 60,000 psi for waste container/sleeve; density 152 pcf for both; and thermal expansion 22.2 X 10(E-6)/degrees F for salt, 6.5 X 10(E-6)/degrees F for waste container/sleeve. The results indicate a horizontal stress of approximately 3300 psi acting on the sleeve, assuming intimate contact between salt and sleeve. Local failures of salt adjacent to the sleeve and in the vicinity of the excavated room are not observed; the model remains elastic throughout the excavation and heating periods. A reduced strength analysis indicates that a salt safety factor of approximately 2 is present. However, it is felt that the tendency of the heated salt to rise would almost certainly induce an upward

<67> CONT.
shearing action along the lateral surface of the sleeve or, if salt/sleeve interface bonding were sufficient, the sleeve would be lifted with the salt. Separation between the sleeve and drillhole bottom could occur, inducing an axial tensile stress in the sleeve. If an initial gap were left between the sleeve and the salt, horizontal loading would be greatly reduced. The resulting heat transfer through the bottom of the sleeve, the associated temperature field induced in the adjacent salt, and their effects on room stability, need to be investigated for accurate assessment of the overall effects. (LKM)

MODELS; MODELS, MATHEMATICAL; SALT DEPOSITS; COMPUTER CODES; EXCAVATION; CONTAINERS; HEAT SOURCES; ELASTOPLASTICITY; THERMAL EXPANSION; STRESS; HEAT TRANSFER; STORAGE, GEOLOGIC; REPOSITORY; STRUCTURAL STABILITY; THEORETICAL STUDIES

<68>
Piper, A.M.

Rock Types, Also Geologic and Hydraulic Settings, Favorable to Deep Placement of High-Level Radioactive Wastes.

Y/OWI/SUB-3785/4; 34 pp. (1978, October)

The objective of this report is to identify those rock types and geologic-hydrologic settings in which a repository for high-level radioactive wastes might be expected to remain functionally intact for at least a thousand, and possibly hundreds of thousands of years, deep beneath the biosphere. A rock medium favorable to deep placement of high-level radwastes should, primarily: 1) be unweathered and distinctly homogeneous (isotropic) in both micro- and macrotextures, over a space sufficient to contain not only the galleries of a repository, but also adequate buffer zones between those galleries and any major surface of discontinuity, such as a stratigraphic boundary or a fracture having a selvage of crushed rock; 2) have minimal intergranular pore space and pore water content, particularly interconnected pore space imparting hydraulic conductivity, and with joints whose width should not greatly exceed 0.1 mm and whose minimum spacing should not be less than 1 m; 3) be non-slaking, for example, contain no anhydrous minerals which might bulk in the presence of moist air. Other rock characteristics that might influence the design or operation of a repository, but ordinarily would not be definitive are: density, mechanical strength, elastoplasticity, thermal conductivity, solubility, potential chemical reactivity between mineral constituents of the rock and waste constituents, and the magnitude and orientation of residual stress in the rock. Seven petrologic types seem most likely to offer masses of rock satisfying the three primary requirements for a waste repository. These are, in general order of diminishing promise: 1) salt, bedded or (possibly) domal; 2) shale or mudstone, non-fissile; 3) granite and other holocrystalline rocks; 4) certain metamorphic rocks - chiefly gneiss, quartzite, and schist if indistinctly foliated; 5) limestone, noncavernous; 6) tuff, dense or welded; and 7) chalk. After an extensive discussion of the geologic and hydrologic settings to be considered in the selection of a repository site, it is concluded that preferable sites would be in upland areas of moderate to slight relief, tectonically stable since the mid-Pleistocene, with the repository horizon below the local base level of erosion. Effective containment of unmodified high-level waste for terms longer than a few thousand years would require a virtually water-free stratum; only bedded, and possibly domal, salt offers such an environment widely. Criteria for a repository site could be relaxed considerably if actinides, nuclides with half-lives over 1000 years, and non-radioactive toxins were removed or rendered insoluble, or if the vicinity of the repository were to be so controlled that the maximum permissible concentrations of the persistently toxic constituents could be relaxed two orders of magnitude from those applied to public drinking water. (Auth) (LKM)

GEOLOGY; SITE SELECTION; REVIEWS; THEORETICAL STUDIES; STORAGE, GEOLOGIC; REPOSITORY; POROSITY; HYDRAULIC CONDUCTIVITY; JOINTS; SALT DEPOSITS; SALT DOMES; SHALES; MUDSTONES; GRANITES; GNEISSES; QUARTZITES; SCHISTS; LIMESTONES; TUFFS; CHALKS; IGNEOUS ROCKS; SEDIMENTARY ROCKS; METAMORPHIC ROCKS; HYDROLOGY; WASTES, HIGH-LEVEL; EVALUATION

<69>
Potter, J.M., Los Alamos Scientific Laboratory, Los Alamos, NM.

Experimental Permeability Studies at Elevated Temperature and Pressure of Granitic Rocks.

LA-7228-T; 101 pp. (1979, May)

Permeability of quartz monzonite from the Los Alamos hot-dry-rock geothermal well GT-2 and Westerly granite was measured experimentally as a function of temperature and pressure. Permeability of the GT-2 rocks from depths of 8580 ft and 9522 ft behaves like Westerly granite for changes in effective confining pressure. As temperature is increased, the permeability of Westerly granite passes through a slight minimum and then increases exponentially above 100 C. Upon cooling the permeability shows a permanent increase of up to four times its original value. The permeability of GT-2-9522, on the other hand, drops off exponentially with increasing temperature, reaching a minimum near 100 C; above 150 C, permeability rises slowly. The GT-2-8580 sample shows somewhat similar behavior, although the permeability shows a minimum near 130 C, followed by a rather sharp rise in permeability at higher temperature. These changes in permeability with temperature are postulated to be caused by differential thermal expansion. Temperature disequilibrium of the GT-2 rocks in their present IN SITU environments is believed to have caused the differences in the behavior between the two samples. This disequilibrium may provide a method for determining the pre-intrusion geothermal gradient of the Jemez area. Flow channels were observed in GT-2 samples using Hi 63 as radioactive tracer. These channel-tracing efforts indicate that at low pressures, fluid transport through the rock is predominantly along intergranular channels. Intragranular channels were not observed with this technique. (Auth) (RAF)

Permeability; Thermal Expansion

<69>

CHEMICAL AND PHYSICAL ASPECTS

<69> CONT.
GRANITES; PERMEABILITY; ROCKS; TEMPERATURE; THERMAL PROPERTIES; ANISOTROPY; ROCK MECHANICS; NICKEL 63; SODIUM 22; SULFUR 35; TRACERS; HEAT ENERGY; LABORATORY STUDIES; EQUATIONS; CORES; VISCOSITY; THERMAL EXPANSION; HYDRAULIC FRACTURING

<70>
Bai, B., R.J. Serne, D.A. Moore, and R.W. Stromatt, Battelle-Pacific Northwest Laboratories, Richland, WA; Hanford Engineering Development Laboratory, Richland, WA.

Electron Transfer Method of Controlling Eh During Adsorption of Multivalent Elements by Geologic Media.

BNL-NB-6766; 17 pp.(1978, October)

The study presents the usefulness of a potentiostat in controlling Eh of soil suspensions and in maintaining Pu(+6) in solution during adsorption by geologic media. It was necessary to determine if this method was adequate in order to study adsorption and migration of multivalent elements in the geologic media where their distribution is highly dependent on their oxidation states in groundwater. An electrolysis cell capable of withstanding high centrifugal forces was used. A preliminary run indicated that the potentiostat was capable of maintaining Pu(+6). For the experiments kaolinite, montmorillonite, gibbsite, and vesicular basalt were used. Results show that the oxidation state distribution did not change with time and essentially all Pu was present as Pu(+6). The data indicate that the potentiostat rigidly controlled Eh as well as it maintained Pu(+6) in solution during its adsorption by soil minerals. (NDV)

ADSORPTION; BASALTS; Eh; GIBBSITE; LABORATORY STUDIES; KAOLINITE; MONTMORILLONITE; PLUTONIUM; VALENCE; OXIDATION; CHEMICAL PROPERTIES

<71>
Relyea, J.F., B. Bai, and R.J. Serne, Battelle-Pacific Northwest Laboratories, Richland, WA.

Interaction of Waste Radionuclides with Geomedia: Program Approach and Progress.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 379-394), 563 pp. (1979)

Current methodologies of studying radionuclide-geologic media interactions are presented. The need for reproducible sorption Kd's and/or migration rates is great, so an effort is underway to determine the most accurate methods. At present there are many studies involved in computer modelling formulation in areas such as groundwater movement, geochemical interactions, terrestrial and aquatic food chains, radiation dose to man, geologic stability and disruptive event analysis. There is a lack of sorption-desorption data and this limits the above models' credibility. Sorption-desorption data is being generated from statistical methods for trends. For validation purposes laboratory experiments will be conducted. Also reviewed are additional field studies, larger lysimeter, or intact core laboratory studies. (NDV)

METHODS; WASTE-ROCK INTERACTIONS; SORPTION; DISTRIBUTION COEFFICIENT; MODELS; REVIEWS; STATISTICS; LABORATORY STUDIES; FIELD STUDIES

<72>
Roedder, E., and H.E. Belkin, U.S. Geological Survey, Reston, VA.

Application of Studies of Fluid Inclusions in Permian Salado Salt, New Mexico, to Problems of Siting the Waste Isolation Pilot Plant.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 313-329), 563 pp. (1979)

This paper presents a summary of the data obtained when fluid inclusions in Permian Salado Salt in cores from the proposed Waste Isolation Pilot Plant site (for radioactive wastes) were examined petrographically, and with microscope freezing, heating, and crushing stages. Fluid now present in the samples studied ranges widely from <0.1 to 1.7 wt %; the total in situ fluid content of the rock is certainly greater. The inclusion fluid is not just saturated NaCl solution but a variable and extremely saline bittern containing some noncondensable gases under pressure. The inclusions move in thermal gradients and react quickly to uniaxial stress, but decrepitation is estimated to be a more important mechanism of fluid release under thermal loading. Subsequent movement of the released fluids is not considered. Too many factors appear to seriously affect the rate of movement of inclusions to permit useful theoretical modeling of the effects of the thermal loading of a canister at this time -- such as: composition of the brine; inclusion size; vapor/liquid ratio; noncondensable gas pressure; growth defects and solid inclusions; growth-induced or later tectonically or thermally induced strain in the host crystal; ambient temperature; temperature gradient; grain size; nature, width, and continuity of grain boundaries; possibility of organic films on boundaries disrupting continuity; and surface tension forces at the bubble/liquid interface. Eight conclusions are listed. (Auth) (DCW)

Thirteen photographs of fluid inclusions are provided in ten figures.

WASTES, RADIOACTIVE; WASTE DISPOSAL; WASTE STORAGE; LABORATORY STUDIES; EXPERIMENTS; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; GASES; CORES; METHODS; TEMPERATURE; PRESSURE EFFECTS; STRESS; THERMAL PROPERTIES; WIPP; BRINES; SALT DEPOSITS; FLUID INCLUSIONS; DECREPITATION; GEOLOGIC FORMATIONS

<73>
Sandia Laboratories, Albuquerque, NM.

Geological Characterization Report, Waste

<73> CONT.
Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Special Studies of Waste Isolation Pilot Plant (WIPP) Rocks.

SAND-78-1596; Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, D.W. Powers, et al. (Eds.), Vol. 2, Ch. 9, (pp. 9-1 - 9-85), 616 pp. (1978, December)

Special studies are being conducted to aid in evaluation of the WIPP site for isolation of radioactive wastes. Rocks from the site have been tested in 4 broad areas: (1) petrography, (2) physical properties (density, moisture content, resistivity), (3) thermomechanical properties (quasi-static and creep parameters), and (4) radionuclide sorption. Physical properties representative of rock salt found in the WIPP horizons are summarized as follows: density 2.18 g/cm cu; porosity 0.5% (range 0.1-0.8%); moisture loss to 300 C 0.4 wt% (range 0-1.0 wt%); resistivity 50,100 ohm-meters (range 6930-230,000); gas permeability less than 3.05 X 10(E-6) darcy; compression wave velocity 4.5 km/sec (range 4.02-4.62); thermal conductivity 5.75 W/mK. Quasi-static properties measured included unconfined strength, secant modulus, principle strain ratio, strain at failure for various confining pressures, tensile strength, and initial yield stress. Results analyzed to date indicate that WIPP salt may undergo both transient and steady-state creep. Steady-state creep rates appear to range from 10(E-10) to 10(E-7)/sec. Transient creep results indicate that temperature has the greatest effects on creep rate. A survey of the potential of geological media from the vicinity of the WIPP site in southeast New Mexico for retardation of radionuclide migration via an aqueous carrier has been completed. Solution simulants representing water in equilibrium with potash minerals or halite zones and a typical groundwater were spiked with radionuclides and contacted with geological samples to obtain sorption coefficients (Kd's). The samples were taken from potential aquifers above and below the repository horizons and from bedded salt deposits in the repository horizons. The nuclides chosen for study included Cs 137, Sr 85, I 131, Tc 99, Sb 125, Ce 144, Ru 152, Gd 153, Ru 106, Am 243, Cm 244, and Pu 238. In brine solutions, Tc and I were not significantly absorbed by any of the minerals and Cs and Sr showed minimal adsorption (Kd's less than 1). The lanthanide and actinide Kd's were typically greater than 10(E+3) and Ru and Sb Kd's varied in the range 25 to over 10(E+3). In the groundwater simulant, Tc and I showed the same behavior, but the Kd's of the other nuclides were generally higher. Some initial parametric studies involving pH, trace organic constituents in the simulant solutions, and radionuclide concentrations were carried out. Differences in the observed Kd's can result from varying one or more of these solution parameters. The WIPP site rocks, including rock salt, show an affinity for radionuclide sorption. Even small values of Kd (less than 1) are effective in retarding movement of radionuclides in groundwaters. (Auth) (LKH)

See also SAND 78-0297, Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico. Permeability, thermal conductivity, and stress-strain plots and tables of rock properties and radionuclide distribution coefficients are included at the end of the chapter.

Creep Rate; Compressional Wave Velocity; Density; Thermal Conductivity; Resistivity; Sorption Coefficient

Cs 137; Sr 85; I 131; Tc 99; Sb 125; Ce 144; Ru 152; Gd 153; Ru 106; Am 243; Cm 244; Pu 238

WIPP; WASTE-ROCK INTERACTIONS; RADIONUCLIDE MIGRATION; GROUND WATER; BRINES; RADIONUCLIDES; SORPTION COEFFICIENT; ROCKS; EVAPORITES; SALTS; HALITE; POTASH; AQUIFERS; SITE SELECTION; SITE EVALUATION; SALT DEPOSITS; PHYSICAL PROPERTIES; THERMAL PROPERTIES; DENSITY; POROSITY; RESISTIVITY; PERMEABILITY; THERMAL CONDUCTIVITY; STRAIN; CREEP; LABORATORY STUDIES

<78>
Sandia Laboratories, Albuquerque, NM.

Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Geochemistry.

SAND-78-1596; Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, D.W. Powers, et al. (Eds.), Vol. 2, Ch. 7, (pp. 7-1 - 7-109), 616 pp. (1978, December)

This chapter summarizes the findings of geochemical investigations of the evaporites and associated non-evaporites of the Delaware Basin in the vicinity of the proposed WIPP site. The most common mineral in the Permian (Ochoan) Salado Formation, which is proposed for radwaste emplacement, is halite, some of which contains minor amounts of anhydrite, and traces of trioctahedral clays and detrital minerals. Marker beds of anhydrite and polyhalite occur locally at intervals of a few tens of meters in the Salado Formation. Local accumulations of potassium and magnesium sulfates occur in the McNutt potash zone in the upper Salado. Petrographic textures, geochronology, and the presence of minerals which are not primary precipitates of seawater, such as sylvite and polyhalite, indicate that much of the evaporite section last underwent recrystallization more than 200 million years ago, shortly after Permian deposition. The trace clay mineral fraction has efficiently entrapped and held radiogenic Sr since the Permian. The aqueous solutions involved in this recrystallization were not in communication with fluids outside the evaporite section. Volatiles presently occur in the evaporites as chemically-combined water of crystallization, hydroxyl groups bound in sheet silicates, short-lived seeps exposed in mines, and as physically entrapped inter- and intracrystalline fluid inclusions. Volatile recovery from halite heated to 600 C typically amounts to less than 0.5 wt% (about 1.5 vol%). In addition to water, traces of N2, CO2, and contaminants from the coring operation are recovered from subsurface evaporite cores. The fluid inclusions appear to have been entrapped at 20-45 C, and contain NaCl, MgCl2, and minor amounts of other solutes; those not affected by recrystallization have not moved in the ambient geothermal gradient since the Permian. Isolated pockets of aqueous solutions now found in the evaporites have no geochemical relationship to surface-derived meteoric waters or to groundwaters above and below the evaporites; neither do they represent evaporite mother liquors. They probably are relics of the post-depositional

<74>

CHEMICAL AND PHYSICAL ASPECTS

<74> CONT.
recrystallization which took place over 200 million years ago, at which time they may have inherited their geochemically distinct solute assemblages and stable isotope compositions. Stable isotope and solute content studies of meteorically-derived groundwaters west of the WIPP site and peripheral to the Delaware Basin indicate that simple uptake of solutes from rocks with which they have come in contact did not alter their meteoric isotope ratios. Thus, waters participating in active dissolution of salt, which has not occurred in the Salado Formation at the WIPP site, could be readily identified by their geochemistry. The origin and age of an artesian brine pocket in the Castile Formation northwest of the WIPP site was evaluated by a uranium isotope disequilibrium model. It was concluded that this accumulation (encountered in borehole ERDA No. 6) has no connection with any other known groundwaters, and has been in its present environment for at least 880,000 years. (Auth) (LKH)

Photographs, charts, and extensive tables of data are included at the end of the chapter.

GEOCHEMISTRY; WIPP; SITE EVALUATION; EVAPORITES; HALITE; POLYHALITE; SILVITE; CLAYS; ANHYDRITE; POTASH; SULFATES; GROUND WATER; BRINES; AGE DETERMINATION; LABORATORY STUDIES

<75>
Science Applications, Inc., Oak Ridge, TN; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Technical Support for Generic Environmental Impact Statement: Radioactive Waste Isolation in Geologic Formations. Volume 19 - Thermal Analyses.

Y/OWI/TM-36/19; 260 pp.(1978, April)

This volume discusses the thermal impacts of isolated high-level and spent fuel wastes in geologic formations. A detailed account of the methodologies employed is given as well as selected results of the analyses. The analyses reported result from three different waste types from alternative LWR fuel cycles: reprocessed waste resulting from uranium and plutonium recycle (Total Recycle or Cycle I); spent unreprocessed reactor fuel (SURF or Cycle II); and reprocessed waste resulting from the recycle of uranium only (U-only Recycle or Cycle III). A single-level repository for these wastes is assumed to lie approximately 2000 ft below the surface in a geologic formation of one of several generic types: domed or bedded salt; granite (crystalline); shale (argillaceous); or basalt (volcanic). The temperature fields resulting from emplacement of these wastes will have consequences in three regions: (1) the near-field region encompassing the storage rooms and pillars between the storage rooms; (2) the very-near-field, which includes the immediate vicinity and the interior of the waste canisters; and (3) the far-field, that is, at distances much removed from the sources. The major intent of these studies is to determine the thermal loading limits based upon criteria imposed in the near-field, very-near-field, and far-field zones. Maximal areal loadings (kW/acre), for HLW and spent fuel (5-yr retrievability) and for spent fuel (25-yr retrievability), based on near-field thermomechanical stress studies are: salt = 150 and 36; granite = 190 and 80; shale = 120 and 55; and basalt = 190 and 80. Maximum canister loadings (kW/can), based on very-near-field studies are: based on limiting canister temperatures for vitrified waste in salt, 3.2; in granite, 1.7; in shale, 1.2; in basalt, 1.3; for calcined waste in salt, 2.6; in granite, 1.6; in shale, 1.1; in basalt, 1.1; based on limiting waste temperatures for vitrified waste in salt, 3.6; in granite, 2.3; in shale, 1.6; in basalt, 1.9; for calcined waste in salt, 2.9; in granite, 2.0; in shale, 1.5; and in basalt, 1.7. Plots of temperature versus time at various depths below the surface have been generated from the far field analyses. The concept of removing heat from the repository by means of ventilation was investigated in relation to the 25-yr retrievable option. The analysis indicates that free convective heat transfer will be sufficient to remove virtually all the heat generated by the canisters to the storage room air stream within a period of 2 years after emplacement. The ventilation required to prevent room closure over the 25-yr retrievable period was also investigated. The feasibility of the concept depends heavily on the ability of the ventilating air stream to absorb sizeable portions of the generated heat without the requirement of inordinate quantities of ventilation. Analysis has shown that sufficient quantities of heat can be removed with air flow velocities that are hardly perceivable, and without the need for specially designed canister sleeves to facilitate heat flow. (Auth) (LKH)

A supplement entitled "Refined Convective Heat Removal Report", by Parsons, Brinckerhoff, Quade and Douglas, Inc., is included in the abstract.

Thermal Load

THEORETICAL STUDIES; THERMAL PROPERTIES; SPENT FUELS; WASTES, HIGH-LEVEL; STORAGE, GEOLOGIC; GRANITES; SHALES; BASALTS; SALT DEPOSITS; SALT DOMES; IGNEOUS ROCKS; SEDIMENTARY ROCKS; THERMAL LOAD; TEMPERATURE; HEAT SOURCES; HEAT TRANSFER; VENTILATION; DESIGN

<76>
Smith, D.D., Oak Ridge Y-12 Plant, Oak Ridge, TN.

Thermophysical Properties of Conasauga Shale.

Y-2161; 27 pp.(1978, December 6)

Thermo-physical property measurements of live Conasauga shale cores representative of eastern illitic shale found under the Oak Ridge reservation were determined at temperatures between 298 and 673 K. Methods of specimen fabrication for different tests were evaluated. Thermal-conductivity and thermal-expansion data were found to be dependent on the structure and orientation of the individual specimens. Thermal conductivities ranged between 2.8 and 1.0 W/m-K with a small negative temperature dependence. Thermal expansions were between 2 and 5 x 10(E-3) over the temperature range for the group. Heat capacity varied with the composition. (Auth) (RAT)

Photographs and photomicrographs of tested materials are given.

Thermal Conductivity; Thermal Expansion; Heat

<76> CONT.
Capacity

SITE SELECTION; LABORATORY STUDIES; THERMAL PROPERTIES; SHALES; THERMAL CONDUCTIVITY; THERMAL EXPANSION; METHODS; MEASUREMENTS; PHYSICAL PROPERTIES; HEAT CAPACITY

<77>
Smyth, J.R., R.M. Crowe, and P.M. Hallock, Los Alamos Scientific Laboratory, Los Alamos, NM.

An Evaluation of the Storage of Radioactive Waste within Silicic Pyroclastic Rocks.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978, (p. 26), 77 pp. (1978)

The occurrences and the hydrological, thermal, mechanical, and mineralogical properties of silicic pyroclastic rocks (tuff) are compared with those of salt, granite, shale, and basalt. Great thicknesses tuff occur at the Nevada Test Site. The average permeability for zeolitized tuffs is 6 X 10(E-8) darcys and for clayey tuffs is 6 X 10(E-6) darcys. Such permeabilities compare favorably with other rock types. A low density tuff has a somewhat lower conductivity than shale at 20 C, while a welded tuff approaches that of granite (3.0 w/mk). Fertilized tuffs are superior to other rock types due to very high sorptive abilities. Nisitg bedded tuff is comparable to salt. Due to their overall characteristics, tuffs, salts, granites, shales, and basalts should be considered for a high-level radwaste repository. (NDV)

This paper was presented at the conference but was not published in the Proceedings.

Permeability

TUFFS; THERMAL PROPERTIES; SALTS; GRANITES; SHALES; BASALTS; ZEOLITES; HYDROLOGY; GEOLOGY; MINERALOGY; PERMEABILITY; DENSITY; SORPTION; REPOSITORY; WASTES, HIGH-LEVEL; ROCKS

<78>
Spitsyn, V.I., V.D. Balukova, S.A. Kabakchi, and N.I. Medvedova, Institute of Physical Chemistry, Academy of Sciences of the USSR, Moscow, USSR.

Influence of Radiation on the System Liquid Radioactive Wastes - Geologic Formation.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 249-255), 563 pp. (1979)

Introduction of liquid radioactive wastes into deep geologic strata results in a number of physical-chemical processes: precipitation, dissolution, complex formation, sorption, etc. Factors that influence the waste-rock system are radioactivity (which leads to radiolysis and heat generation), chemical composition of waste, and waste-rock interactions under radiation and pressure at depth. The area occupied by the injected waste and changes in the nature of the liquid phase depend primarily on radiolysis processes in the heterogeneous system of liquid waste-geologic material occurring at elevated temperatures and pressures. Experiments that simulate actual conditions of temperature, pressure, and high radiation levels on this system have been performed. Results are presented for radiolytic gas formation (such as hydrogen, carbon dioxide, and nitrogen) and for changes in the liquid phase and sorption capacity of host rock minerals. It is shown that the temperature increase in the geologic strata significantly enhances waste decomposition processes, promotes sorption of radionuclides, and decreases the mobility of the waste in the formation. Six kinds or groups of experiments are described: a) an experiment on radiolysis kinetics of nitric acid at different dose powers and irradiation temperatures; b) experiments in which aqueous solutions of sodium nitrate and nitric acid were exposed to a range of temperatures and pressures with and without the addition of certain metals; c) an experiment showing how the quantity and kinetics of metal ion precipitation depend on many factors, including temperature, the initial acid concentration, heating time, etc.; d) several experiments to explore waste-host rock interactions at elevated temperatures and pressures; e) an experiment in which carbonate host material in contact with liquid waste was irradiated to show the kinetics of nitrogen and hydrogen evolution and oxygen consumption; and f) experiments on the influence of radiation on minerals and their sorption capacities, which included investigation of irradiated minerals by X-ray and infra-red spectrometry. (Auth) (DCW)

This paper is largely concerned with physical chemistry experiments on liquid radioactive waste-rock interactions. Two figures present data on radiolysis kinetics of nitric acid at different dose power and irradiation temperatures, and nitrogen and oxygen consumption during thermoradiolysis of a "waste-strata" system.

pH; Temperature

WASTE STORAGE; WASTE DISPOSAL; WASTES, RADIOACTIVE; WASTES, LIQUID; WASTE-ROCK INTERACTIONS; STORAGE, GEOLOGIC; HOST ROCKS; GEOLOGIC STRATA; GEOLOGIC FORMATIONS; RADIOLYSIS; DECAY HEAT; EXPERIMENTS; TEMPERATURE; PRESSURE EFFECTS; RADIATION EFFECTS; RADIOACTIVE DECAY; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; CHEMISTRY; LABORATORY STUDIES; METHODS; MINERALS; CHEMICAL COMPOSITION; PRECIPITATION, CHEMICAL; COMPLEXES; IONIC PROCESSES; KINETICS; GASES; SORPTION

<79>
Stewart, D.B., U.S. Geological Survey, Branch of Experimental Geochemistry and Mineralogy, Reston, VA.

United States Geological Survey Studies of Physical-Chemical Relationships in Salt Repositories.

Y/OWI/SUB-77/18268; CONF-770789; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 65-67), 92 pp. (1977, August 20)

The amount and physical properties of brine that can occur in salt repositories at

<79>

CHEMICAL AND PHYSICAL ASPECTS

<79> CONT.
elevated temperatures and pressures adjacent to waste-bearing canisters will have considerable impact on the strength and stability of the repository. Brine will form readily from H2O absorbed on surfaces, diffusion along grain boundaries, movement of fluid inclusions, dehydration of hydrous minerals such as gypsum, polyhalite, or clays, or even from leakage through failed shaft openings. The temperature-pressure diagram for NaCl-H2O describes the limits for coexisting solid-liquid-gas assemblages in salt repositories. Isobaric T-X diagrams show compositional details at pressures below and above the critical point (about 440 C). Addition of components causes shrinkage of the solid-liquid-vapor loop on these diagrams; equilibrium is attained rapidly, large freezing point depressions occur, and the boundary of the liquid field shifts markedly. This shift is being determined for the system NaCl-CaSO4-H2O and for the phase volume in the system NaCl-KCl-MgCl2-CaCl2-CaSO4-H2O that is in equilibrium with solid NaCl and CaSO4 (or intervening compounds). All of the physical properties of the ensuing bulk brines will be measured, or deduced from a corresponding state argument and properties of the system NaCl-H2O. Because aggregates of solids are mechanically weakened by interstitial liquid, determination of mechanical properties of brine-bearing aggregates is needed. A maximum of 1/3 by weight of brine, and probably much less, will destroy rock strength. Hence, physical chemical data will set limits to permissible H2O access at specified temperatures, assist in the design of critical measurements of mechanical strength, and will enable the volume and other properties of brines generated by the access of H2O to be determined. (Auth) (LEW)

Only the abstract is provided.

BRINES; SALT DEPOSITS; REPOSITORY; STORAGE, GEOLOGIC; THERMAL PROPERTIES; PRESSURE EFFECTS; PHYSICAL PROPERTIES; STRENGTH, MECHANICAL; LABORATORY STUDIES; THEORETICAL STUDIES; EVAPORITES; WASTE-ROCK INTERACTIONS

<80>
Stewart, D.B., and R.W. Potter, Jr., U.S. Geological Survey, Reston, VA; U.S. Geological Survey, Menlo Park, CA.

Application of Physical Chemistry of Fluids in Rock Salt at Elevated Temperature and Pressure to Repositories for Radioactive Waste.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 297-311), 563 pp. (1979)

Salt deposits usually have less than 1% H2O within the salt mass. Water may occur in fluid conclusions as bittern and as brine composed of Ca and Mg chlorides. Water will also be present in such hydrous minerals as polyhalite, carnallite, tachyhydrite, bischofite, gypsum, or hydrous silicate minerals. This fluid migrates when mechanical stress is imposed, such as when a repository perturbs a salt deposit mechanically, thermally, and chemically. The fluid released could accumulate near the waste making it at best less recoverable and at worst might compromise the repository and its seals. To better design waste repositories the salt-brine-waste canister physical chemistry must be understood. The vapor pressure at repository temperature of CaCl2 or MgCl2 bittern is less than lithostatic pressure so no gas phase is expected. However, corrosion will be severe in the presence of such bittern. Also the mechanical strength of salt formations is reduced when bittern are present. More data are being obtained in laboratory and field experiments. (NSW)

BRINES; CHEMICAL PROPERTIES; CONTAINERS; CORROSION; FLUID MECHANICS; MINERALS; PHYSICAL PROPERTIES; REPOSITORY; ROCK MECHANICS; WASTE-ROCK INTERACTIONS; SALT DEPOSITS; THERMAL PROPERTIES

<81>
Swyler, K.J., R.W. Klaffky, and S.W. Levy, Brookhaven National Laboratory, Upton, NY.

Radiation Damage Studies on Natural and Synthetic Rock Salt for Waste Disposal Applications.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 349-354), 563 pp. (1979)

Appreciable radiation damage is to be expected in the rock salt adjacent to radioactive waste disposal canisters in salt deposits. In sufficient concentrations, the defects formed will modify the physical and chemical properties of the rock salt. Defect formation in natural rock salt is similar but not identical to defect formation in synthetic NaCl. Experimental studies of radiation damage were done using optical absorption measurements on samples held at controlled temperature, during and after 1.5 MeV electron irradiation. Natural rock salt and synthetic NaCl develop negative ion defects (F centers) when irradiated between 20 and 300 C. Both form high concentrations of sodium metal colloid particles when irradiated in the 125 to 200 C region. In synthetic NaCl maximum colloid formation occurs in the 150 to 180 C range. In both materials the F-center saturation levels vary with temperature and dose rate but are remarkably independent of plastic deformation or geological origin. Based on current data, it is estimated that the exposures expected from high-level waste disposal canisters will cause 0.01 to 0.1 mole percent of the sodium in the surrounding rock salt to be converted to sodium metal colloid particles. Defect formation is dependent on temperature, crystal strain, dose, dose rate, origin of crystals, and possibly other factors. (Auth) (DCW)

Three figures illustrate absorption spectra during and after electron bombardment of natural rock salt, growth of F center and colloid absorption during irradiation of synthetic NaCl and natural rock salt, and growth of colloid absorption during irradiation of plastically deformed synthetic NaCl. A table gives radiation damage in natural rock salt at 150 C.

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTE-ROCK INTERACTIONS; WASTE STORAGE; WASTE DISPOSAL; STORAGE, GEOLOGIC; CANISTERS; SALT

<81> CONT.
DEPOSITS; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; CHEMISTRY; IRRADIATION; SODIUM; GEOLOGIC DEPOSITS; DOSE RATE; TEMPERATURE; LABORATORY STUDIES

SALTS; HALITE; SALT DEPOSITS; REPOSITORY; CANISTERS; RADIATION EFFECTS; IRRADIATION; LABORATORY STUDIES; THERMAL PROPERTIES; STRAIN; COLLOIDS; KINETICS; WASTE-ROCK INTERACTIONS; WASTES, RADIOACTIVE

<82>
Seyler, R.J., and R.W. Levy, Brookhaven National Laboratory, Upton, Long Island, NY.

Radiation Damage Studies on Natural and Synthetic Rock Salt Utilizing Measurements Made During Electron Irradiation.

I/OSI/SUB-77/14268; CONF-770749; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 69-71), 92 pp. (1977, August 20)

The radiation damage effects which will occur in rock salt surrounding radioactive waste disposal canisters are being investigated with an apparatus for making optical and other measurements during 1-3 MeV electron irradiation. This equipment consists of a computer controlled double beam spectrophotometer which simultaneously records 256 point absorption and radioluminescence spectra, in either the 200-400 or 500-800 nm region, every 40 sec. This procedure provides information on the kinetics and other details of the damage formation process and when the radiation is terminated, on both the transient and stable damage components. The exposure rates may be varied between 10(E+2) or 10(E+3) to more than 10(E+6) rad/hr and the sample temperature maintained between 25 and 900 C. Measurements have been made on synthetic NaCl and on natural halite from two potential disposal sites and two mines (including samples from AEC drill hole #7 on the WIPP site). Both unstrained and purposely strained samples have been used. Most recently, measurements at temperatures between 25 and 200 C have been started. Measurements to date indicate that the damage formation kinetics in natural halite are quite different from those observed in synthetic NaCl. At room temperature the F-center versus irradiation time curves differ appreciably. The halite curves do not contain the prominent first stage exhibited by synthetic NaCl but do contain a square root of irradiation time component which is not usually observed in synthetic alkali halides other than LiF. At room temperature the F-center growth curves continue to increase at large doses and little or no colloid formation is observed. In contrast, the F-center growth curves observed at higher temperatures indicate that the F-center concentrations reach a saturation level. Also, at these temperatures, extensive colloid band formation is observed. When room temperature irradiations are terminated transient effects occur in both synthetic and natural crystals. Transients, other than those with half-lives of an hour or more, are not seen at higher temperatures. Long-term annealing effects have not been observed in natural samples between 25 and 200 C. However, long-term annealing has been observed in moderately irradiated synthetic samples at 200 C. This suggests that the back-reactions which limit F-center formation in natural samples at elevated temperatures operate only during irradiation. (Auth) (LKR)

Only the abstract is given.

<83>
Thompson, J. L., and E. Wolfsberg, Los Alamos Scientific Laboratory, Los Alamos, NM.

Applicability of Microautoradiography to Sorption Studies.

LA-7609-MS; 12 pp. (1979, January)

The technique of microautoradiography was applied to the study of the sorption of U and Am on five rock types which exist at the Nevada Test Site. Autoradiograms were prepared in relatively short times using modest levels of activity. Solutions of U 233 or Am 241 with activity of 10(E+6) to 10(E+7) dis/min/ml contacted with rock for a few hours gave sufficient transfer of activity to produce good quality radiograms in exposure times not exceeding a few days. The density of tracks in the radiogram can be directly correlated with the tendency of a mineral to sorb the radioactive species, and particularly good sorption sites are clearly indicated by stars in the developed emulsion. The extent of sorption is not quantitatively measured but it provides insight into the sorption process because the state of aggregation of the sorbed particles is indicated. (RAP)

SORPTION; AUTORADIOGRAPHY; LABORATORY STUDIES; URANIUM 233; AMERICIUM 241; ROCKS; MINERALS; ALLUVIUM; GRANITES; TUFFS; ALPHA PARTICLES; ARGILLITES

<84>
Tonnessen, R.B., and J.J. Cohen, Lawrence Livermore Laboratory, Livermore, CA.

Survey of Naturally Occurring Hazardous Materials in Deep Geologic Formations: A Perspective on the Relative Hazard of Deep Burial of Nuclear Wastes.

UCRL-52199; 13 pp. (1977, January 14)

Hazards associated with deep burial of solidified nuclear waste are considered with reference to toxic elements in naturally occurring ore deposits. The elements referred to are As, Ba, Cd, Cr, Pb, Se, Ag, Th, and U. The basis for comparison derives from a consideration of safe drinking water levels. Calculations for relative toxicity of fast breeder reactor (FBR) waste and light water reactor (LWR) waste in an underground repository are compared with the relative toxicity indices obtained for average concentration ore deposits. Results indicate that, over time, nuclear waste toxicity decreases to levels below those of naturally occurring hazardous materials. (Auth) (RAP)

The relative toxicity of nuclear waste over time compared with that of average mineral ores of toxic elements is illustrated.

THEORETICAL STUDIES; BURIAL; WASTES, RADIOACTIVE; GEOLOGIC FORMATIONS; HAZARD

<84>

CHEMICAL AND PHYSICAL ASPECTS

<84> CONT.
ANALYSIS; TOXICITY; ORES; EVALUATION; REPOSITORY; DRINKING WATER; STANDARDS, FEDERAL; REACTORS, BREEDER; REACTORS, LIGHT WATER; ARSENIC; BARIUM; CADMIUM; CHROMIUM; MERCURY; LEAD; SELENIUM; SILVER; URANIUM 235; URANIUM 238; THORIUM 232; SOLUBILITY; ORE DEPOSITS; GEOLOGIC DEPOSITS

<85>
Voight, B., Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Rock Stresses Associated with Burial of Nuclear Waste.

T/ONI/SUB-77/18268; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (p. 73), 92 pp. (1977, August 20)

Rock stress changes related to long-term deep storage of nuclear waste involve thermoelastic and pore fluid pressure changes associated with excavation and heating. We are now examining computer models to assess the question of thermally-induced fracturing in storage rock surrounding radioactive waste containers. Stresses are evaluated in three dimensions, employing elastic-plastic finite element codes. Potential failure conditions are expressed in terms of "effective stresses", and force and thermal fields are incremented to produce an appropriate load path. In general, heating in the vicinity of waste containers produces a zone of high compression bonded by a zone of circumferential and axial tension. Fractures are possible in radial directions as well as perpendicular to the axis of the cylindrical heat source. In addition, the mechanical effect of a vapor pulse will be explored by a two-phase numerical fluid transport model used in conjunction with mechanical finite element models. A tensile pulse is expected to propagate out from the site along with the temperature field. In addition, interaction will occur as the temperature pulses of different canisters approach each other and superimpose. This will intensify the stress, and tension fractures may be expected. Later, the compressive pulse may close these fractures if mineralization has not occurred. Preliminary work suggests the possibility of establishing design criteria (design burial depth, depth of canister below storage vault, etc.) in order to minimize problems of potential rock fracture. (Auth) (LKM)

Only the abstract is given.

MODELS; MODELS, MATHEMATICAL; THEORETICAL STUDIES; COMPUTER CODES; STORAGE, GEOLOGIC; THERMAL PROPERTIES; HYDRAULICS; EXCAVATION; ROCK MECHANICS; STRESS; STRAIN; CONTAINERS; HEAT SOURCES; HEAT TRANSFER; FRACTURES; DESIGN; WASTES, RADIOACTIVE; WASTE-ROCK INTERACTIONS

<86>
Neweik, W.R., and D.W. Hanson, Sandia Laboratories, Albuquerque, NM.

Mechanical Behavior of New Mexico Rock Salt in Triaxial Compression up to 200 C.

SAND-77-119.2C; CONF-780564; Salt, Proceedings of an International Symposium, Hamburg, German Federal Republic, May 29, 1978. (24 pp.). (1978)

An extensive experimental program is being conducted to determine the mechanical behavior of New Mexico rock salt in support of the structural design of a Radioactive Waste Isolation Pilot Plant (WIPP). Three groups of tests are discussed to identify the relative and site-specific importance of deviator stress, confining pressure (mean stress), temperature, time (loading rate), and stress path. The three groups of experiments consist of (1) hydrostatic loading, (2) conventional triaxial compression tests (sigma sub 1 greater than sigma sub 2 or equal to sigma sub 3 equal to const.), and (3) variable stress path tests including experiments at approximately constant sigma sub 1 and at constant mean stress. All data were generated on 100 mm diameter specimens. The rock salt exhibited nonlinear response under all loading conditions, practically zero initial elastic limit and an apparent inseparability of permanent deformations into mm-independent and time-dependent components. Pressure and temperature did not alter the elastic constants but affected the principal strain ratio, the ratio volumetric strain/shear strain, rock salt ductility, and the ultimate stress. In particular, low pressure and temperature permitted pronounced dilatancy and loss in load bearing ability. Under such conditions the volumetric strains reach sizable fractions of the shear strains. Pressure remained important even at high temperature because it influenced the rate of shearing. Load path and stress history may be significant under deviatoric loading conditions and for large variations in pressure. (Auth) (RAP)

Figure 1 gives description of experimental apparatus. Figures 2-6 give stress-strain curves for New Mexico rock salt under varying conditions.

WIPP; LABORATORY STUDIES; SALT DEPOSITS; STRESS; TEMPERATURE; TIME FACTOR; ELASTOPLASTICITY; STRAIN; PRESSURE EFFECTS; ROCK MECHANICS

<87>
Weaver, C.E., Georgia Institute of Technology, School of Geophysical Sciences, Atlanta, GA.

Geothermometry Methods for Determining the Thermal History of Shales.

T/ONI/SUB-77/18268; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 75-76), 92 pp. (1977, August 20)

Geothermometry studies are being conducted to allow prediction of diagenetic changes that will occur when shales are exposed to the heat generated by radioactive waste. Initial studies are of the geothermal history of the Conasauga Shale near Oak Ridge, Tennessee. Other study sites in Tennessee, northwest Georgia, and eastern Alabama will also be investigated. The study involves investigation of outcrop, quarry, and core samples. Detailed chemical and mineralogic studies of the various size fractions will be made to determine the variation in the make-up of the clay suite, the illite and chlorite polytypes, crystallinity index, 10

<87> CONT.

A/S A peak-height ratio, Na-K feldspar ratios, etc. Criteria will be developed to determine the extent and nature of the diagenetic changes that have occurred as a function of temperature and depth of burial. Scanning and transmission electron microscopes will be used to observe morphologyical, textural, and structural changes that are related to temperature. Potassium-argon studies will be made of the illitic material in the various size fractions, and the variation in K/Ar ratio as a function of temperature and solubility will be measured using the Ar 40/Ar 39 method, to determine the extent to which K and Ar have been mobilized. Radiogenic argon present in the pore waters of the relatively impermeable shales will be measured. Oxygen 18/oxygen 16 analyses will be made of two or more coexisting minerals, probably quartz and illite, from various size fractions of the samples. If the two minerals have reached isotopic equilibrium, the maximum temperature to which they have been subjected can be calculated. Vitrinite reflectance measurements will be made to determine the level of organic metamorphism. These values along with visual kerogen assessment will be used to estimate the maximum burial temperature. The effective heating time is a significant factor and makes the organic metamorphism-temperature relation somewhat subjective; however, with K-Ar, O 18/O 16, mineralogic data it should be possible to obtain time-temperature information that can be used to make the organic temperature scale fairly accurate. (Auth) (LKH)

Only the abstract is given.

THERMAL PROPERTIES; SHALES; SEDIMENTARY ROCKS; CORES; OMTCPOPS; GEOCHEMISTRY; MINERALOGY; ILLITE; CHLORITES; FELDSPARS; DIAGENESIS;; WATER; ISOTOPE RATIOS; ORGANIC COMPOUNDS; FIELD STUDIES; LABORATORY STUDIES; WASTE-ROCK INTERACTIONS

<88>

White, A.F., and H.C. Claassen, U.S. Geological Survey, Nuclear Hydrology Program, Denver, CO.

Kinetic Model for Silicate Rock-Ground Water Interaction.

Y/OWI/SUB-77/10268; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 77-78), 92 pp.(1977, August 20)

Experimental and theoretical evaluation of kinetic reaction mechanisms and rates that control aqueous interaction with silicate phases is being conducted by the Geological Survey. Present emphasis is on dissolution and hydrolysis of vitric tuffs from the Nevada Test Site, a proposed repository of high-level nuclear waste. Dissolution experiments have been conducted over pH ranges of 0.5-7.5, temperatures of 12-50 C, and glass surface to aqueous volume ratios of 35 sq m/l-300 sq m/l. The experiments used a rhyolitic tuff (72% SiO2) from Yucca Mesa and laboratory water modified to simulate natural waters. Devitrification phases in the tuff yielded calcium and bicarbonate species to the solution while the glassy phases yielded sodium and bicarbonate species. It has been found that pH is related to the valence state of the ions and therefore the parabolic diffusion rate is strongly dependent upon the divalent calcium ion concentration. The parabolic diffusion mechanisms can be explained by solid state counter-ion diffusion between hydrogen ions and species contained within the glass structure. The pH dependence of the reaction constant is proportional to the valence states of the diffusing species. The temperature dependence can be described by the Arrhenius equation. A substantial fraction of the total mass transferred from the solid is absorbed at the glass/solution interface. Numerical solutions to the diffusion equations can describe surface to volume effects and solution concentrations on the reaction rate by utilizing a Freundlich sorption isotherm which relates solution concentrations to glass-surface concentrations. Higher surface concentrations effectively decrease the gradients present in the solid phase which in turn decreases the diffusion flux and therefore the reaction rate. Extrapolation of the experimental reaction rates to a long time interval successfully duplicates the ionic composition of ground water associated with tuffs at NTS. Future modifications are planned for the model that will quantitatively predict pH and temperature effects on the reaction rates of individual species. Coupled with hydrologic information, the model should successfully estimate the history and fate of chemical species introduced both from the glass phase and from artifical sources. (Auth) (LKH)

Only the abstract is given.

SITE SELECTION; WASTE-ROCK INTERACTIONS; DISSOLUTION; KINETICS; SILICATES; HYDROLYSIS; TUFFS; REPOSITORY; STORAGE, GEOLOGIC; LABORATORY STUDIES; SITE SELECTION; CALCIUM; SODIUM; pH; VALENCE; DIFFUSION; GLASS; THERMAL PROPERTIES; GROUND WATER; MODELS; MODELS, MATHEMATICAL; WASTES, RADIOACTIVE; THEORETICAL STUDIES

<89>

White, W.B., Pennsylvania State University, Materials Research Laboratory and Department of Geosciences, University Park, PA.

Waste-Rock Interactions and Bedrock Reactions.

Y/OWI/SUB-77/10268; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 79-80), 92 pp.(1977, August 20)

This experimental program is designed to discover possible reactions between shale repository rocks and radioactive wastes. The canister can be regarded in three ways: (a) as simply a source of heat that modifies the mineralogy and therefore the physical properties of the surrounding rock (dry heat); (b) as a source of heat that activates reactions between minerals in the surrounding rock and slowly percolating ground water; or (c) as a source of reaction materials of different composition from the surrounding rock and which therefore may react to form completely new "minerals" in a contact aureole around the canister. The matrix of waste-rock interactions contains two composition axes. One is the composition of the shales of which the items whose variability are of most interest are sulfides (which may act as sources of acid), organic materials (which may generate a highly

<89> CONT.

reducing environment, low Eh, in which heavy metals tend to be more mobile), and carbonates (which may release gas on decomposition, and which act to buffer the pH of any circulating solutions). The second axis is waste composition, defined by the various prototype waste forms usually investigated: glass, calcine, "spent fuel" and the ceramic supercalcine. This latter is composed of well-defined crystalline phases and these could be investigated individually. In all interactions water plays a deciding role. The temperatures and pressures at which these reactions take place must be investigated. Experiments at higher pressure and temperature allow reactions to take place on a laboratory time scale and thus identify what could happen. These reactions are then followed downward in temperature to determine both phase boundaries and kinetic cut-offs below which equilibrium cannot be achieved on a laboratory time scale. The experiments described above emphasize the solid phases and their reactions, regarding the fluid phase as merely a medium for enhancing reaction rates. An alternate set of experiments now only in the design stage uses the circulating fluid phase as a means of investigating reactions within the host rocks and between the host rock and the waste material. In the circulation system, temperature flow rate, pressure head, and water composition can be controlled directly. By controlling the porosity of the material placed in the chamber, the flow hydraulics can also be controlled to some extent. The uptake of solute into the circulating aqueous solution and changes in pH and Eh of the solution provide alternate sources of information about reactions among the solid phases in the chamber. (Auth) (LKW)

Only the abstract is given.

SHALES; SEDIMENTARY ROCKS; CONTAINERS; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; HEAT SOURCES; HEAT TRANSFER; PHYSICAL PROPERTIES; THERMAL PROPERTIES; GROUND WATER; HYDROTHERMAL ALTERATIONS; CHEMICAL PROPERTIES; THEORETICAL STUDIES; pH; Eh; ORGANIC COMPOUNDS; PRESSURE EFFECTS; LABORATORY STUDIES; WASTE-ROCK INTERACTIONS

<90>
Witherspoon, P.A., C.H. Amich, J.E. Gale, and K. Iwai, University of California, Department of Materials Science and Mineral Engineering, Berkeley, CA; Lawrence Berkeley Laboratory, Berkeley, CA.

Observations of a Potential Size-Effect in Experimental Determination of the Hydraulic Properties of Fractures.

LBL-8571; SAC-17; 15 pp. (1979, May)

In several recent investigations, experimental studies on the effect of normal stress on the hydraulic conductivity of a single fracture were made on three rock specimens ranging in cross-sectional area from 0.02 sq m to over 1.0 sq m. At the maximum stress levels that could be attained (10 to 20 MPa), minimum values of the fracture hydraulic conductivity were not the same for each rock specimen. These minimum values increased with specimen size indicating that the determination of fracture conductivity may be significantly influenced by a size effect. The implications of these results are important. Cores collected in the field are normally not larger than 0.15 m in diameter. However, the results of this work suggest that when this size core is used for laboratory investigations, the results may be non-conservative in that fracture permeabilities will be significantly lower than will be found in the field. If the results with an ultralarge core (0.95-m diameter) are more indicative of field conductivities for a fracture under stress, then further work is needed to determine optimum specimen size so that reliable results on flow in fractures under stress will be available. (Auth)

Hydraulic Velocity

FRACTURES; LABORATORY STUDIES; HYDRAULIC CONDUCTIVITY; ROCKS; STRESS; CORES; PERMEABILITY; EQUATIONS; HYDRAULIC HEAD; ROCK MECHANICS

<91>
Derwoth, A., Los Alamos Scientific Laboratory, Los Alamos, NM.

Gas Generation from Radiolytic Attack of Transuranic-Contaminated Hydrogenous Waste.

LA-7678-MS; 26 pp. (1979, June)

The possibility that alpha-radiolysis of hydrogenous materials might produce toxic, corrosive, and flammable gases in retrievable stored waste prompted an investigation of gas identities and generation rates in the laboratory and field. Typical waste mixtures were synthesized and contaminated with Pu 238 for laboratory experiments, and drums of actual transuranic-contaminated waste were instrumented for field testing. Several levels of contamination were studied, as well as pressure, temperature, and moisture effects. Gas values were determined for various waste matrices, and degradation products were examined. Studies of several years' records for contaminated waste stored at Los Alamos Scientific Laboratory and also at Idaho National Engineering Laboratory indicate that drum contents average less than 10 gm of Pu 239 per 210 L drum, and many drums contain 1 gm or less. Such drums were found not to generate measureable quantities of gas. Experiments showed that elevated temperature increases the rate of gas formation; however, this effect appears to diminish with time. The rate of gas formation decreases as pressure increases; the possible benefit in case of deep underground disposal is obvious. If, as the data seem to indicate, this phenomenon is the result of recombination of H2 and O2, then the rate of water formation under pressure would be greater initially than the 10.5 mg of H2O/kg of cellulosic waste per year observed. However, it would diminish as H2 production diminished and a source of O2 was no longer available. Extrapolation of this curve shows that the diminution of water production probably would take place during the working life of a repository; for the long term, the gas and water production would be insignificant. The amount of gas generation from polyethylene was found to be very similar to that from cellulosics. However, the gas composition is significantly different: polyethylene produces almost all H2. (Auth) (LKW)

Pu 239

<91> CONT.
HYDROLYSIS; ALPHA PARTICLES; GASES; LABORATORY STUDIES; FIELD STUDIES; WASTES, ORGANIC; NITROGEN; PLUTONIUM 238; PLUTONIUM 239; CELLULOSE; POLYETHYLENE; DRUGS; THERMAL PROPERTIES; PRESSURE EFFECTS; WASTE STORAGE; WASTE DISPOSAL; MOISTURE; WASTES, TRANSURANIC; WATER; WASTES, RADIOACTIVE; CONTAMINATION; TRANSURANICS

<92>
Ad Hoc Committee on the Transportation of Plutonium by Air, National Research Council, Aeronautics and Space Engineering Board, Washington, DC.

Review of Criteria for Packaging Plutonium for Transport by Air.

NUREG/CR-0420; 56 pp. (1978)

In the summer of 1975 the shipment, by air, of plutonium was halted until a suitable container for such transport could be designed. A paper on the proposed qualification criteria for plutonium package certification was generated. Some of the tests recommended for determining the strength of the package are: impact at 422 ft/sec at a right angle onto a flat unyielding surface; static compressive load of 70,000 lbs; package to be exposed to flames of aviation fuel for at least 60 minutes; and immersion in at least 3 ft of water for 8 hours. These tests are to be done in series. Individual tests are free fall impact and deep submersion. Acceptance standards are: Containment (no vessel is ruptured), shielding (external radiation not exceeding 1 Rem/hr at a distance of 3 ft), subcriticality (a package or an array of packages must demonstrate subcriticality). The plutonium packages are to be placed in the aft-most cargo section, securely cradled and tied down, shipped with no other hazardous materials, and be fitted with Emergency Locator Transmitters. It is expected that this report will result in a packaged container that will not rupture in the crash and explosion of a high-flying aircraft. (NTW)

Appendix includes the procedure for calculating quantities of plutonium. An attachment is provided describing the compliance certificate.

ACCIDENTS; CONTAINER INTEGRITY; CONTAINERS; DESIGN; HAZARD ANALYSIS; IMPACT SHOCK; LEGISLATION; LICENSING; WASTES, TRANSURANIC; TRANSPORTATION, AIR; PLUTONIUM

<93>
Adams, P.B., Corning Glass Works, Corning, NY.

Chemistry of Nuclear Waste Glass Reactions: Problems and Potential of Prediction.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 123-129), 563 pp. (1979)

Since it is impossible to do real-time experiments in determining release of radioactivity from a storage system, extrapolations must be made. However, the validity of such extrapolations depends on a clear perception of various pitfalls resulting from the composition of the glasses employed. Factors such as temperature, pH, silica solubility, and dislodgement of reaction layers affect the reaction rates. Some assumptions are not always accurate, for instance: a comparison is a hot test ranks glasses in the same order as in a cold test where the slope of reaction versus temperature may differ; solubility increases with increasing temperature with no accounting for excess salts; and solvent chemistry remains constant through the test. Reaction rates are strongly pH dependent, and failure to consider pH will cause incorrect interpretations. Implications of the unconsidered factors are discussed. (NTW)

PREDICTIONS; pH; SILICA; TEMPERATURE; CHEMICAL PROPERTIES; GLASS

<94>
Andrews, L.J., Chem-Nuclear Systems, Inc., Bellevue, WA.

Engineered Storage for the Disposal of Non-Fuel Bearing Components from Nuclear Facilities and General Radioactive Waste.

IAEA-SM-207/70; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, (5 pp.)-(1976, March)

Chem-Nuclear Systems, Inc. has developed a unique Non-Fuel Bearing Component (NFBC) waste transport cask and cask handling design interfaced with an optimum safe NFBC waste storage vault plan. The unloading tasks have been improved to control exposure better by handling the waste containers below grade in slit type trenches 20 ft deep. A plan has been developed to reduce to a minimum the personnel exposure conditions and involves an improved cask design and improvements in site burial techniques. A top-loading bottom-unloading transport cask and trailer unit designed to interface with an engineered waste disposal system simplifies handling, minimizes exposure, maximizes site utilization and reduces burial site selection problems. The cask/trailer unit, located over a vertical disposal sleeve in a Special Material Sleeve Vault (SMSV), is employed to transfer the canister from the cask to the disposal sleeve in a concrete vault by a release mechanism which prevents the release of airborne radioactive particles and permits cavity air sampling. Detailed drawings of cask and waste canister, combined cask and truck trailer package, engineered storage-site plan, and cask handling cycle are given. (CRJ)

CASKS; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTE DISPOSAL; REPOSITORY; DESIGN; CONTAINERS; DISPOSAL SITE; PARTICLES, AIRBORNE; EXPOSURE, OCCUPATIONAL; FIELD STUDIES; METHODS; STORAGE, GEOLOGIC; SURFACE CONTAMINATION; TRANSPORTATION; TRENCHES; WASTE MANAGEMENT; WASTE STORAGE

<95>
Ingerman, C.L., and W.H. Rankin, Savannah River Laboratory, Aiken, SC.

Durability of Containers for Storing Solidified Radioactive Wastes.

DP-MS-76-66; 15 pp. (1976)

The durability of five alloys (AISI 1020 carbon steel, "Cor-Ten A," Type 304L stainless steel, "Inconel 600", and "Inconel 625") considered for use in containers for the storage of solidified radioactive wastes is discussed in the report. When the cost of materials for the containers is included, 1020 carbon steel appears to be the best

<95>

CONTAINER DESIGN AND PERFORMANCE

<95> CONT.

candidate of the alloys considered. The choice was based on the oxidation resistances, strengths, and compatibilities of the alloys under conditions expected during 100 year storage, melting of glass by either in-can or continuous-melter processes, and impact and thermal accidents, such as loss of cooling and fires. The thinner wall thickness required for satisfactory performance of the stronger, more oxidation-resistant alloys, such as 304L stainless steel is offset by their higher cost per pound. A 0.61 m (2 feet) in diameter by 3 m (10 feet) high carbon steel container with a wall thickness of about 1.3 cm (0.5 inch) would be expected to survive most service and accident environments with either a concrete or glass waste form. In this analysis, survival is conservatively defined as less than 1% deformation by creep during storage, about 1% creep in a loss of cooling accident, no penetration in an accident, and creep of less than 1% (but no rupture) in a fire. Because of the high internal pressures associated with concrete-filled containers in thermal accidents, and the high temperatures associated with in-can melting of glass, much thicker walls (up to 25.4 cm (10 inch)) would be required to survive these conditions. (JHT)

STEELS; ACCIDENTS; CONTAINERS; COST BENEFIT ANALYSIS; ALLOYS; CONTAINER INTEGRITY; CREEP; CONCRETES; GLASS; WASTES, RADIOACTIVE; WASTES, SOLID; LABORATORY STUDIES

<96>

Arnold, C., Jr., Sandia Laboratories, Polymer Science and Engineering Division, Albuquerque, NM.

Evaluation of Organic Coolants for the Transportation of Liquid Metal Fast Breeder Reactor Spent Fuel Rods.

SAND-77-1486; 39 pp. (1978, May)

The physical and chemical processes that are likely to occur when sodium coated LMFBR spent fuel rods are submerged in various aromatic organic coolants was defined by means of immersion experiments carried out with sodium coated 304 stainless steel coupons. Upon immersion of sodium coated coupons at 220 degrees C in hydrocarbon type coolants such as Therminol 66, a mixture of terphenyls, not only was the metallic sodium retained on the coupon, but a carbonaceous coating formed on the surface of the sodium. In contrast, coolants that contained aromatic ether bonds, such as Dowtherm A, reacted with sodium at 220 degrees C to form phenolate and other salts, which precipitated from the coolant in the form of a dark sledge. With Dowtherm A, removal of metallic sodium from the coupon was essentially complete in a matter of hours at temperatures of 160-220 degrees C. Data on the rate and efficiency of sodium removal upon immersion in Dowtherm A at elevated temperatures were obtained. In addition the kinetics and chemistry of the sodium/Dowtherm A reaction were defined. Because sodium sledges are potentially incompatible with the containing structural materials and the fuel elements, it is recommended that sodium be removed prior to immersion in the coolant via reaction with benzoic acid; this method should be adaptable to the facilities at reactor sites. In aging studies Dowtherm A was found to be thermally stable up to 400 degrees C and radioactively stable at ambient conditions. The combined effect of heat and radiation was not defined. (auth)

ACIDS, ORGANIC; CHEMICAL PROPERTIES; COOLANTS; FUEL RODS; LABORATORY STUDIES; PHENOLS; REACTORS, LIQUID METAL FAST BREEDER; SODIUM; SLUDGES; TRANSPORTATION; THERMAL PROPERTIES; HEAT EXCHANGE; SPENT FUELS

<97>

Palu, K., S. Ramachandran, and R.K. Jain, Bhabha Atomic Research Centre, Trombay, Bombay, India.

Management of Highly Active Cladding Zircaloy Solid Wastes from a Fuel Reprocessing Plant.

CONF-7[illegible]: [illegible]/PUB/833; IAEA-SM-207/[illegible]; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 1, (pp. 179-183), 397 pp. (1976)

This paper describes a system for the collection, transport, and disposal of "chop-leached" Zircaloy-2 hull wastes from a fuel reprocessing plant in India. The design of the disposal facility at Tarapur Solid Waste Storage Site takes into account the high radioactivity and long-lived nature of these wastes, and the local geohydrologic conditions. The water table in the area is relatively high (about one meter below the surface) during the monsoon season, but with rapid drainage due to the fractured state of the basaltic bedrock. Thus there is a need for very effective isolation of wastes. Zircaloy wastes are collected in 200-liter mild steel barrels with acid-resistant inner linings. These are stored in underground circular vaults ("tile holes") made to hold three stacked barrels each. The vaults are composed of concrete/steel/concrete layered pipes, water-proofed on the outside, with a welded steel bottom set in concrete and a 1-meter thick concrete shield on top. As a further protection against the pyrophoric nature of Zircaloy wastes, they are designed to be flooded with cement grout after the barrels are in place. Also, the areas between tile holes are paved with concrete. Soil and groundwater monitoring is carried out continuously. This system is considered an interim solution to the waste disposal problem; suitable geologic strata are currently under study for ultimate disposal. (LKR)

Clay Content; Permeability; Porosity

CASKS; CLADDING; CONTAINER INTEGRITY; CONTAINERS; CONTAINMENT; DECLADDING; DESIGN; EQUIPMENT; FUEL REPROCESSING; FUELS; METHODS; STORAGE, GEOLOGIC; TRANSURANICS; CONTAINMENTS; SHIELDED CONTAINERS; GEOLOGY; HYDROLOGY; WASTE DISPOSAL; WASTE MANAGEMENT; WASTE STORAGE; WASTES, SOLID; WASTES, TRANSURANIC; WASTES, HIGH-LEVEL; ZIRCALOYS; PERMEABILITY, INTRINSIC

<98>

Bibler, N.E., Savannah River Laboratory, Aiken, SC.

Radiolytic Gas Production from Concrete

<98> CONT.
Containing Savannah River Plant Waste.

DP-1460; 39 pp. (1978, January)

Samples of concrete and simulated high-level waste were irradiated with Co 60 gamma rays (simulating beta particles from fission products) and Cm 244 alpha particles to determine the amount of radiolytic gas production. Tests were conducted on concrete containing Fe2O3 or MnO2, and mixtures of hydrous oxides of Fe3+ and Al3+ or Fe3+ and Mn4+. Irradiations of 8.9 X 10(E+4) rads/hr, 7.0 X 10(E+7) rads/hr, and 2.0 X 10(E+7) rads/hr were made. Hydrogen was the only major product from gamma radiolysis, reaching a steady-state pressure that increased with radiation intensity. Nitrous oxide was also formed when the waste contained nitrates or nitrites. Oxygen was depleted. Gas volume and water-cement ratios did not significantly affect the steady-state H2 pressure, which was 96-100 psi for a dose rate of 2.0 X 10(E+7) rads/hr. Irradiating samples of concrete containing Fe-Al waste and ascorbic acid as an organic set retarder resulted in faster H2 production and a higher steady-state pressure [152 psi using 7.0 X 10(E+7) rads/hr]. Tests with actual SRP waste in concrete achieved a steady-state pressure of 7 psi at a dose rate of 3 X 10(E+4) rads/hr; an initial pressure decrease to below atmospheric was due to O2 depletion. It is estimated that initial dose rates in concrete waste forms will be 5.0 X 10(E+4) rads/hr from beta-gamma and 300 rads/hr from alpha waste. During the first 300 yrs of storage, Cs 137 and Sr 90 decay should produce steady-state H2 pressures of 8-20 psi and O2 pressures near zero; maximum N2O pressures will be under 60 psi. Pu 238 and Pu 239 decay will then increase H2 to over 200 psi. Final pressures in containers 2 ft in diameter X 10 ft with 10% free volume will be about 1000 psi H2 and 700 psi O2. (LKM)

RADIOLYSIS; CONCRETES; GASES; PREDICTIONS; ALPHA PARTICLES; RADIATION, GAMMA; IRRADIATION; CONTAINMENT; WASTE STORAGE; WASTES, HIGH-LEVEL; HYDROGEN; OXYGEN; NITROGEN COMPOUNDS

<99>
Bibler, N.E., and J.H. Kelley, Savannah River Laboratory, Aiken, SC.

Effect of Internal Alpha Radiation on Borosilicate Glass Containing Savannah River Plant Waste.

DP-1402; 16 pp. (1978, May)

Effects of internal alpha radiation on borosilicate glass, a prospective matrix for long-term storage of Savannah River Plant (SRP) radioactive waste, were evaluated in samples containing 85 wt % simulated waste [Fe(OH)3-MnO2] and either 0.5 wt % Cm 244 or 1 wt % Pu 238. A glass containing Pu 238 without simulated waste was also studied for comparison. The glasses were examined for changes in physical stability, leachability, and dilatation. Alpha dose rates in the test glasses ranged from 9.5 X 10(E+18) to 1.3 X 10(E+19) alpha dis/(g-day). After 420 days, microcracks had formed; however, no macrostructural damage to the glasses was observed. Leachabilities for Cm 244 and Pu 238 were <7 X 10(E-8)/g (2 sq cm-day) and were not affected by the radiation. Continuous leaching by water for 5 days removed <10(E-5)% of the isotopes. Fractions leached were 2 X 10(E-8)% for Cm 244 and 2 X 10(E-7)% for Pu 238, indicating that Pu is not as strongly bound in the matrix. Expansion rates in vol %/g rad were 0.017 for Pu 238 glass with waste, 0.012 for Cm 244 glass with waste, and 0.0083 for Pu 238 glass without waste. For glass containing SRP waste the dose after 10(E+6) years will result in a fractional expansion of less than 0.1%. It is calculated that stored energy will be less than 6 cal/g, which could not cause a temperature rise of more than 50 deg. C in the glass, insufficient to cause damage. (Auth) (LKM)

Leaching Rate; Thermal Expansion

Cm 244; Pu 238

DOSE RATE; ALPHA PARTICLES; GLASS; LEACHING; TEMPERATURE; PREDICTIONS; CONTAINMENT; WASTE STORAGE; WASTES, HIGH-LEVEL; RADIATION EFFECTS; LABORATORY STUDIES

<100>
Bibler, N.E., and E.C. Ostteugh, Savannah River Laboratory, Aiken, SC.

Radiolytic Gas Production from Tritiated Waste Forms.

DP-1459; 28 pp. (1977, July)

Radiolytic gas production during long-term storage of tritiated waste was estimated from gamma and alpha radiolysis tests to determine the extent of pressurization in sealed containers. Two forms of simulated wastes were irradiated with Co 60 gamma rays or Cm 244 alpha particles: concrete for solidification of tritiated water and vermiculite for solidification of tritiated octane or vacuum pump oil. Results of these test irradiations were used to estimate the effects of beta radiolysis. For concrete, the gamma and alpha radiolysis results predicted that H2 will be formed by tritium beta particles with an initial rate of 0.1 to 0.3 molecule for every 100 eV of energy absorbed. Also, as the H2 pressure increases, this 100-eV yield decreases because of a reaction removing H2. Eventually, a steady state pressure that depends on the radiation intensity will be attained. For intensities less than 10(E+5) rads/hr, the steady state pressure will be less than 20 psi. O2 in the air sealed with the concrete will be almost completely depleted, and H2 will be unaffected. For the organic materials sorbed onto vermiculite, the gamma and alpha radiolysis results predicted that H2 and traces of CH4 and CO2 will be produced. For tritium beta particles, the 100-eV yields for H2 based on energy sorbed by the organic materials are 0.8 for octane and 2.2 for vacuum pump oil. In the containers, steady state H2 pressure will not be attained at pressures up to at least 200 psi. As with the concrete, O2 will be nearly completely depleted and H2 will be unaffected. The 100-eV yield for H2 production was used to calculate pressure increases in conceptual tritiated waste packages. For a 27 gal drum containing 10(E+5) Ci of H 3 solidified in 20 gal of concrete with a void volume of 10%, initial H2 pressurization will be 0.08-0.12 psi/day at 23 C; a steady state pressure of less than 10 psi is predicted. The same drum containing 10(E+8) Ci of H 3 in 7.0 gal of

<100> CONT.
octane or vacuum pump oil sorbed onto 20 gal of vermiculite would initially pressurize at 0.044 and 0.02 psi/day, respectively, for octane and oil; final pressures would be 310 psi for octane and 155 psi for vacuum pump oil. (Auth) (LEN)

RADIOLYSIS; CONCRETES; VERMICULITE; TRITIUM; WATER; ORGANIC COMPOUNDS; GASES; PREDICTIONS; ALPHA PARTICLES; RADIATION, GAMMA; IRRADIATION; CONTAINMENT; WASTES, LIQUID; WASTE STORAGE; RADIOLYSIS

<101>
Bradley, R.J., Battelle-Pacific Northwest Laboratories, Richland, WA.

History of Prototype High Level Waste Canister SS-9 While in Air and Water Storage.

PNL-2278; 34 pp. (1977, November)

The history of a leak in a Waste Solidification Engineering Prototype (WSEP) high-level waste canister while in simulated water basin storage was investigated. The canister (SS-9), a prototype fabricated of 310 stainless steel, was filled with a high-level phosphate ceramic waste material containing 2 x 10(E+6) curies of radioactivity. Following 1.2 years water storage at 50 degree C, 3.5 years hot air storage at 400 to 500 degree C, and 10 months water storage at 50 degree C, the canister failed. The canister had three visible cracks, all within the bottom 2 feet of the canister, and one of which was 13 in. long. Coefficients of thermal expansion were 6 x 10(E-5) in./in. degree C and 14.4 x 10(E-6) in./in. degree C for the phosphate glass and 310 steel canister respectively. Thus as the canister cooled upon completion of the run, it became highly stressed since the glass prevented its normal contraction. This evaluation showed that 310 stainless steel becomes highly sensitized after prolonged air storage at elevated temperatures and should not be subsequently stored in water. It was concluded from metallography that failure was due to stress-assisted intergranular attack enhanced by metal sensitization during the hot air storage period, and a high chloride ion concentration in the canister storage water. The predominant isotope in the SS-9 canister the first 4.6 years was Ce 144. After this time, Cs 137 became the major radiation source. The primary objectives of the run was to determine short-term decay effects, since there were no actinides in the waste. Coupons were taken from the canister and the leach rate of the material in deionized water was determined to be 5.1 x 10(E-4) g/sq cm-day for the first day. Averaged over 90 days, the material leach rate was 3.1 x 10(E-5) g/sq cm-day. Since it was known that the failure occurred sometime between quarterly canister storage water sampling periods, these leach rates were used to calculate an effective waste surface area presented by the canister cracks. Thus, the leach rates for the first day and the average for 90 days represent the extremes that could have occurred. The effective waste surface area contacted via the canister cracks was calculated to be between 1 and 20 sq m. Based on the calculated effective surface area and the above leach rates of the phosphate ceramic material, 6.1 g of waste were dissolved per day. (EIJ)

Leaching Rate

Cs 137; Ce 144

WASTE STORAGE; WASTES, HIGH-LEVEL; CONTAINER INTEGRITY; CONTAINERS; CONTAINMENT; HEAT RESISTANCE; LEAKS; TEMPERATURE; STAINLESS STEELS; METALS; WASTES, SOLID; CESIUM; CERIUM; CERAMICS; WASTES, RADIOACTIVE; SAFETY; LABORATORY STUDIES

<102>
Braithwaite, J.W., and R.J. Wagnani, Sandia Laboratories, Albuquerque, NM.

Corrosion Considerations for Nuclear Waste Isolation Canisters.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, C.J. McCarthy (ED.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 283-287), 563 pp. (1979)

Disposal in a stable geologic formation is presently the most viable method for insuring isolation of nuclear wastes. Migration and dispersal of radionuclides into the biosphere is blocked by barriers such as the waste form, the encapsulating metal canister, and the geologic formation. At Sandia Laboratories three geologic formations are being investigated: bedded salt in southeast New Mexico for transuranic defense wastes, unreprocessed fuel rods, and high-level commercial wastes; and sub-seabed formations in the North Pacific and hardrock in Nevada for commercial high-level wastes. The impact of radiolysis alterations on waste form and canister stability must be assessed further. The high-level waste bedded salt and sub-seabed sediment environments might be especially corrosive (high temperature aqueous chloride solutions), and may be more damaging than the internal corrosion associated with the waste. In a site-specific compatibility study at Sandia, corrosion rate determinations for canisters have been done for bedded salt and subsea sediments. Over 20 metal alloys are being considered as candidate canister materials. The possibility of using very thin overpacks and submerged arc deposition or plasma-spraying of expensive alloys onto less expensive substrates is also being considered. In results so far, the effect of gamma radiation on hydrothermal environments and the associated change in the corrosivity of the chloride solutions have not yet been adequately quantified. The effect of intense gamma radiation on waste form durability and metal corrosion at 40 C in brine and seawater is presently being investigated. Experimental results on typical corrosion rates in simulated high-level waste isolation environments lead to several conclusions: a) deoxygenated saturated brine is more corrosive than deoxygenated seawater plus sediments, b) the addition of substantial amounts of dissolved oxygen to the solutions (to simulate the worst-case effect of brine radiolysis) greatly increased the corrosion rate of all alloys tested, and c) localized crevice corrosion and pitting in deaerated 200 C and 250 C seawater and brine was minimal even for a susceptible alloy, but in the presence of oxygen, significant localized attack occurred even in very corrosion resistant alloys. In-situ testing and laboratory experimental results have led to

<102> CONT.

the conclusion that long lifetime canisters can be provided, and that corrosion resistant alloys such as Hastelloy C-276 and Ticode 12 are the most cost effective. (Auth) (DCW)

Table 1 provides temperature, pressure, and chemistry descriptions of three disposal environments for different waste types; Table 2 lists corrosion rates at 250 C for 10 candidate overpack alloys in simulated environments.

Corrosion Rate

WASTES, RADIOACTIVE; WASTE DISPOSAL; WASTE STORAGE; WASTES, DEFENSE; WASTES, HIGH-LEVEL; WASTES, COMMERCIAL; WASTES, TRANSURANIC; STORAGE, GEOLOGIC; CANISTERS; RADIOLYSIS; SALT DEPOSITS; GEOLOGIC FORMATIONS; SEDIMENTS; SEA DISPOSAL; SITE SELECTION; CORROSION LABORATORY STUDIES; CHEMICAL PROPERTIES; CHEMISTRY; DRIBBS; ALLOYS; REQUIREMENTS

worst-case, brine inundated, 250 C oxidizing environments. The most expensive (per unit wt) alloys appear to be cost effective for both the 25- and 300-yr lifetime alternatives. Alloy requirements can be satisfied with present U.S. production capacity. Based on these initial results, published data, and alloy costs and availability, eight prime candidate alloys have been selected for further comprehensive study: 1019 mild steel, Corten A steel, lead, 90-10 Cupro-nickel, SS-White 26-1, Monel 400, Inconel 600, and Ticode-12. (Auth) (LKH)

Temperature; pH; Corrosion Rate

LABORATORY STUDIES; SALT DEPOSITS; SALT DOMES; SEA DISPOSAL; STORAGE, GEOLOGIC; WASTE DISPOSAL; WASTE STORAGE; CONTAINERS; BRINES; pH; SEAWATER; GROUND WATER; ALLOYS; STEELS; STAINLESS STEELS; OXIDATION; ECONOMICS; COST BENEFIT ANALYSIS; SEDIMENTS; TEMPERATURE; PRESSURE EFFECTS; CORROSION; RADIOLYSIS; OVERPACKING

<103>

Braithwaite, J.W., and M.A. Molecke, Sandia Laboratories, Albuquerque, NM.

High-Level Waste Canister Corrosion Studies Pertinent to Geologic Isolation.

SAND-78-2111C; CONF-781201; Proceedings of a Conference on High-Level Radioactive Solid Waste Forms, Denver, CO, December 19-21, 1978, (31 pp.). (1978, December)

The compatibility of candidate rad waste canister materials with deep geologic isolation environments is being investigated in a series of interrelated laboratory, bench-scale, and in site experiments. Results presented are partially applicable to bedded and domed salt, seabed sediment, and various types of hardrock geologic waste repository or test facilities. Emphasis in this work is on bedded-salt and seabed environments. Several series of laboratory tests were conducted to evaluate the effects of environmental parameters on resultant corrosion rates. The parameters represent both optimized emplacement conditions and credible, worst-case (e.g., water intrusion) boundary conditions. Variables tested include: temperature (25-250 C), pressure, solution composition, oxygen concentrations, solution volume/surface area ratios, welding and crevice effects, and dry and moist salt effects. Approximately 20 candidate metals were initially tested for corrosion rates under appropriate simulated HLW isolation conditions. It was found that test solution corrosiveness increased in the order Brine (NaCl) - seawater - Brine A (Mg/K/Na chloride). Corrosiveness was probably due to a pH effect; the pH of seawater and Brine A decreased from 8.1 and 6.5 at 25 C as the temperature was raised (quenched, 250 C pH values were 3.9a and 3.4, respectively). Brine B pH values changed less than one unit. Localized attack in deaerated 200 C and 250 C test solutions was minimal even for 300L SS. In the presence of oxygen, however, significant localized attack was observed even in the very corrosion resistant alloys such as Carpenter 20Cb3 and Hastelloy C-276. Effects of gamma radiolysis products are still under investigation. The economic aspects of utilizing several candidate canister and canister overpack materials were also addressed. Overpack costs were calculated for possible 1-, 25-, and 300-yr design lifetime requirements under

<104>

Bezzhova, N.E., A.A. Hinaev, and S.B. Ozirnner, Academy of Sciences of the USSR, Institute of Physical Chemistry, Moscow, USSR.

Vitrification of High Sodium-Aluminum Wastes: Composition Ranges and Properties.

CONF-791121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 43-53), 563 pp. (1979)

Using glass blocks as storage containers for radwaste is highly reliable, but there are problems in understanding vitrification and devitrification processes. Studies of Na2O-P2O5 oxides of wastes and Na2O-SiO2-B2O3-CaO oxides of wastes show that three valent oxides act both as glass formators (in phosphate systems) and as modificators. The two valent oxides act only as modificators shifting the range of vitrificaton to the lower content of sodium oxide. When the content of modificators increase the chemical stability lessens. In the process of glass melting the volatility of some components of the wastes, such as Cs and Ru takes place. The glass of high-level fixation for radwastes that do not require high temperatures was accomplished. It was also suggested by the experiment that the leachability of radionuclides increases along the Cs-Sr-Co-Ru progression, and is 10(E-6), 10(E-7), 10(E-8), and 10(E-9) g/sq cm day, respectively. Studies of crystallization of glasses and metamictization are discussed. (NDV)

GLASS; VITRIFICATION; LABORATORY STUDIES; CRYSTALLIZATION; CHEMICAL PROPERTIES; WASTES, RADIOACTIVE; LEACHING; TEMPERATURE

<105>

Cerny, P., University of Manitoba, Department of Earth Sciences, Winnipeg, Manitoba, Canada.

Pollucite and Its Alteration on Geological Occurrences and in Deep-Burial Radioactive Waste Disposal.

<105>

CONTAINER DESIGN AND PERFORMANCE

<105> CONT.
CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 231-236), 563 pp. (1979)

Pollucite belongs to the solid-solution series $NaAlSi_2O_6 \cdot H_2O$-$CsAlSi_2O_6$. Isotropic, cubic members cover the range $Pollucite_{0}$-$Pollucite_{2}$; anisotropic members are low-symmetry compositions in the Analrich range. Minor components are Fe, Rb, K, Ca, Mg, Li, Mn, Tl, Ga, and Sr. Pollucite occurs in the central parts of highly fractionated granitic pegmatites which are enriched in Li, Rb, and F. Crystallization of the mineral occurs in late stage pegmatite consolidation and forms with petalite or spodumene, K-feldspar, and quartz. Pollucite is easily altered, with feldspars, lepidolite, spodumene, muscovite, quartz, montmorillonite, kaolinite, and carbonates are the most common products. Cesium is not retained during alteration and becomes widely dispersed. When considering synthetic pollucite for Cs 137 fixation the following is important: incorporation of Cs is much higher than in the natural form; rarity and specific paragenesis of natural pollucite are caused more by geochemical scarcity and migration pattern of Cs rather than restricted stability; the pressure-temperature ranges of alteration products needs more study; and an extremely steep chemical gradient can be expected between the Cs 137-bearing storage-pollucite and the clarke-level Cs contents of surrounding rocks and solutions. (WDV)

POLLUCITE; MINERALOGY; CHEMICAL PROPERTIES; GEOCHEMISTRY; CRYSTALLIZATION; MINERALS; CESIUM; TEMPERATURE

<106>
Cole, W.A., United Kingdom Atomic Energy Authority Research Group, Atomic Energy Research Establishment, Electronics and Applied Physics Division, Harwell, Berkshire, England.

An Automatic Drum Scanning System for the Measurement of Plutonium in Waste.

Nuclear Instruments and Methods 65:85-57. (1968)

A description is given for an apparatus for locating and estimating the plutonium content of waste drums, without opening them. One is made of an automatic drum scanner and a gamma radiation detector assembly consisting of a NaI(Tl) crystal and a photomultiplier tube. The drum tube examined is positioned horizontally within the scanner, which then rotates the drum about its axis and scans its length with the detector assembly. A single channel analyser accepts the pulses derived from the detector and selects those whose energies lie within a 50 keV channel centered on 380 keV. Compensation of the presence of interfering radiation from other nuclides is possible by use of a separate discriminator, and a second scaler. Minimum detectable quantities of Pu on the waste-drum scanner, for the large detector assembly are 22 mg for slow scan, 45 mg for medium scan, and 105 mg for fast scan. The scanning times are 15 min, 5 min, and 1 min respectively. This method of scanning simulates the condition of 4 pi geometry to within about 8%, and the detection sensitivity is such that plutonium contents within the range 22 mg-20 g may be detected with plus or minus 10% accuracy at the 2 sigma (sigma) level. The scanner may be used equally well for determining the presence of many other gamma emitting isotopes, for example U 235 and I 131. Four waste-drum scanners have been produced, and all are in regular use in various UKAEA establishments. (PTO)

DRUMS; LOGGING, GAMMA; PLUTONIUM; EQUIPMENT; MONITORING; RADIATION, GAMMA; RADIATION DETECTORS

<107>
Cooper, K.C., and F.C. Prenger, Los Alamos Scientific Laboratory, Los Alamos, NM.

Heat Pipe Cooling System for Underground, Radioactive Waste Storage Tanks.

LA-8230-MS; 18 pp. (1980, February)

The objective of this program is to determine the technical and economic feasibility of an air thermosyphon for cooling an underground waste storage tank with a 60,000-100,000 BTU/hr heat generation rate. This is the final report on Task 1 of the program -- the performance of a thermal and energy balance analysis on an underground waste storage tank to determine thermosyphon feasibility, and a parametric study to determine the lowest sludge surface temperature and duct size for the air flow. Task 2 in the program requires a conceptual design report including cost estimates, prototype test unit design, and environmental assessment; Task 3 will analyze prototype test data to scale up the thermosyphon system for an actual tank. The results of the analysis show that an array of 37 heat pipes inserted through a central hole at the top of a radioactive waste storage tank will remove 100,000 BTU/hr with a heat sink of 70 F atmospheric air. Heat transfer inside the tank to the heat pipe is by natural convection. Heat rejection to outside air utilizes a blower to force air past the heat pipe condenser. The heat pipe evaporator section is axially finned, and is constructed of stainless steel. The working fluid is ammonia. The finned pipes are individually shrouded and extend 35 ft down into the tank air space. The hot tank air enters the shroud at the top of the tank and flows downward as it is cooled, with the resulting increased density furnishing the pressure difference for circulation. The cooled air discharges at the center of the tank above the sludge surface, flows radially outward, and picks up heat from the radioactive sludge. At the tank wall the heated air rises and then flows inward to complete the cycle. The design philosophy concentrates on using existing technology combined with adequate performance margins with the goal of minimizing development time and reducing cost. The design is based on passive thermal components inside the tank penetration through existing dome ports. In addition, uncertainties about degradation of the heat pipe working fluid in the radiation environment (about 1000 rad/hr) are overcome by including on-site fluid replacement capability. Boundary temperatures were based on worst case conditions. If minimum temperature limits are established, these can be met by controlling fan power and hence thermal conductance to the outside environment. (Auth) (LKM)

DESIGN; COOLANTS; WASTE STORAGE; TANKS; HEAT EXCHANGE; SLUDGES; TEMPERATURE; EVALUATION; HEAT

<107> CONT.
TRANSFER; CONVECTION; EVAPORATION; AMMONIA; PIPES; COOLANTS; DEGRADATION; WASTES, HIGH-LEVEL

<108>
Elliot, M.N., J.N. Grover, and W.H. Hardwick, Atomic Energy Research Establishment, Harwell, England.

Fixation of Radioactive Waste in Glass. Part 2. The Experimental Evaluation of Phosphate and Borosilicate Glasses.

Treatment and Storage of High-Level Radioactive Wastes, Proceedings of a Conference, Vienna, Austria, October 8-12, 1962. International Atomic Energy Agency, Vienna, (pp. 381-396). (1963)

A glass suitable for long-term storage of fission-product wastes must be chosen with regard to the method of storage and reliability of manufacture in a highly radioactive plant. Factors relevant to storage include leach resistance, thermal and radiation stability and the percentage of waste that can be incorporated, while the physical and chemical nature of glass-forming additives together with the melting point and viscosity of the glass relate to the manufacturing process. The relative importance of these factors is discussed. Experimental studies of the borosilicate glasses that have been chosen for pilot-plant work are described, including the effect of radiation doses of up to 10 (E+12) rad on their physical and chemical properties. It was demonstrated that a borosilicate glass containing 25% waste oxides can be manufactured in the Harwell process. This glass would be a very satisfactory medium for storage of fission-product wastes, being almost insoluble in water and this insolubility is not significantly affected by devitrification or even by very high radiation doses. It does not react with the stainless-steel processs vessel even at the high temperatures of its manufacture. The borosilicate system may not be suitable for incorporation of waste oxides to a greater extent than 30% if the present melting-point limitations persist. Other kinds of glasses were examined and their potential value as storage media compared with the borosilicate glasses. Up to 40% waste oxides may be included in many other glass compositions. In particular the phospho-silicate system gives very good glasses, but while it offers advantages in metering and handling the glass-forming additives, this is offset by the corrosive nature of phosphate melts. (Auth) (CAJ)

GLASS; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTE STORAGE; FISSION PRODUCTS; OXIDES; PHOSPHATE GLASS; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; FIELD STUDIES; RADIATION EFFECTS; THERMAL PROPERTIES; CERAMICS; CONTAINMENT; SIPP

<109>
Empson, F.N., R.L. Hosner, and W.F. Schaffer, Oak Ridge National Laboratory, Oak Ridge, TN.

Operating Experience.

ORNL-4555 (Draft); ORNL-CF-76-7-82; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 10, (pp. 10.1-10.26), 822 pp. (1971, April)

This chapter recounts the events of the operation phase of the experiment, including dismantling and removing the equipment, and presents some important results related to equipment performance in handling the fuel element canisters. The results are primarily related to the performance of and experience with the equipment and for handling the fuel element canisters. The techniques used in handling the canisters were successful. (CAJ)

Graphs and tables are included. Sample data forms for operating procedures are also given.

WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; MINES; DISPOSAL SITE; DESIGN; EQUIPMENT; EVALUATION; FIELD STUDIES; CANISTERS; LABORATORY STUDIES; WASTES, SOLID

<110>
Ensminger, D.A., A.A. Sutherland, Jr., S.G. Oston, and N.A. Van Konynenburg, The Analytic Sciences Corporation, Reading, MA; Lawrence Livermore Laboratory, Livermore, CA.

Release of Radionuclides from High-Level Waste During Transportation Fires.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 541-548), 563 pp. (1979)

A probabilistic analysis determining the expected values of population doses from rail or road transportation was performed. The waste consider was solidified high-level waste from spent commercial reactor reprocessing and consisted of spray calcine, fluidized-bed calcine, borosilicate glass, and a special "multi-barrier" waste form. Stainless steel containers hold the waste. Under conditions of fire the cask seal breaches early so the canister determines the threshold of release. Generally corrosion of the canister is the primary point of failure. If the canister fails the release mechanism will depend on the waste form properties, mechanical form of the ruptured cask, the canister and waste. The calculation may be performed by a double integration over time and temperature of the product of the release function and the probability density functions of fire time and fire temperature in the area of the time-temperature where failure can take place. Particle plumes are the configuration the material takes in the air. This method permits safety comparison between waste forms and transportation modes. (RDV)

STATISTICS; SAFETY; CANISTERS; EQUATIONS; THERMAL PROPERTIES; TEMPERATURE; CASKS; TRANSPORTATION; METHODS

<111>
Ewing, R.C., University of New Mexico, Department of Geology, Albuquerque, NM.

<111>

CONTAINER DESIGN AND PERFORMANCE

<111> CONT.
Natural Glasses: Analogues for Radioactive Waste Forms.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, C.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 57-68), 563 pp. (1979)

The long-term effects of alteration and metamictization in the geologic environment is studied using natural analogues. Experiments have shown that the presence of water and the temperature are the most important rate controlling variables. Besides volcanic glasses, impact glasses, tektites and lunar glasses may occur substantially unaltered. For tektites their resistance to alteration is due to their high SiO2 content and low alkali content. While it is inappropriate to make direct comparisons between natural glasses and radwaste glasses, the natural glasses may serve as bench mark for comparing physical properties resulting from certain given chemical properties. This type of comparison should aid in selection of the appropriate waste form. (NDV)

CHEMICAL PROPERTIES; GLASS; CRYSTALLIZATION; LATTICE; IMMOBILIZATION; WASTES, RADIOACTIVE

<112>
Hall, R.J., Battelle-Pacific Northwest Laboratories, Richland, WA.

Development of High-Level Waste Shipping Cask Models.

BNWL-2126; Nuclear Waste Management and Transportation Quarterly Progress Report, April Through June 1976, A.M. Platt, Section 7.0, (p. 7.1), 44 pp. (1976, October)

The objective of this program is to conceptually design and fabricate models of a cask for transporting high-level waste by railroad. Two types of models of a high-level waste shipping cask are to be fabricated in this project: 1) a model of the cask mounted on an HO-gauge rail car (1:87 scale) and 2) a table model of the cask alone for use in portable displays. The design of a conceptual solidified high-level waste shipping cask system was completed, and the 1/10th scale table display model is being fabricated. The cask for the table model will be about 1 ft in diameter by 2 ft long and will include a cutaway section showing the high-level waste canisters inside. The table display model is expected to be ready early in July. A display poster to accompany the model has been prepared to further illustrate the conceptual design and to characterize the solidified high-level waste transported. Fabrication of the HO-gauge cask car system model has been initiated. The concept drawing and sketches used for model shop fabrication are being refined. These drawings will be sent to various ERDA laboratories for review and comment. (Auth) (LKH)

MODELS; DESIGN; FABRICATION; CASKS; TRANSPORTATION, RAIL; WASTES, HIGH-LEVEL; WASTES, SOLID; WASTE MANAGEMENT; THEORETICAL STUDIES

<113>
Hartwig, C.M., Sandia Laboratories, Livermore, CA.

Raman Spectra of Cesium Aluminosilicate Waste Forms.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, C.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 219-225), 563 pp. (1979)

The Raman spectra were used to delineate the structure of two cesium aluminosilicates with formulations similar to pollucite (Cs2O.Al2O3.4SiO2). These minerals are useful for incorporating large quantities of Cs 137 bearing waste. The resonance at 200/cm involves participation of Cs, at 297/cm Al, and at 392/cm Si. Oxygen stretching vibrations are contained near the frequency range of 1000/cm. The band at 970/cm is associated with non-bridging oxygens, the band at 1040/cm with the Al-O bond, and the band at 1120/cm with the Si-O. The spectra show that water is incorporated into pollucite as a monomer, hydrogen bonded to the oxygen cage in which it is located. (NDV)

SPECTROMETRY; SILICATES; ALUMINUM; OXYGEN; POLLUCITE; CESIUM; LATTICE; LABORATORY STUDIES

<114>
Metherlin, S.W., D.A. Baker, C.E. Beyer, S. Mandel, and P.L. Peterson, Battelle-Pacific Northwest Laboratories, Richland, WA.

Evaluation of the Consequences of Light Water Reactor Spent Fuel and Plutonium Shipping Package Loss at Sea.

IAEA-SR-10/19; Design Construction and Testing of Packaging for the Safe Transport of Radioactive Materials, Proceedings of a Symposium, Vienna, Austria, August 23-27, 1976. International Atomic Energy Agency, Vienna, (pp. 315-323). (1976)

This paper is a theoretical study of the effects of the loss to the sea of shipping packages containing LWR spent fuel or plutonium dioxide. The shipping packages considered are a 100-ton mechanically cooled cask in the case of LWR spent fuel and a 57-liter specification 6M container with standard 2H inner containment in the case of PuO2. Scenarios considered are loss of an undamaged package, loss of a fire-damaged package, loss on the continental shelf (200 m or less), and loss beyond the shelf. Estimated average individual dose and dose commitment from the marine food chain from loss of a Pu package (2.55 kg Pu) on the continental shelf are 4.6 x 10(E-6) rem and 1.1 x 10(E-4) rem, respectively. Estimated average individual dose commitments from loss of a spent fuel cask (3.1 tons U) are 6.3 x 10(E-8) rem for an undamaged cask on the continental shelf, 0.11 rem for an extensively fire-damaged cask on the continental shelf, and less than 1.1 x 10(E-8) rem for a cask (damaged or undamaged) in deep ocean. (LKH)

ACCIDENTS; FIRE HAZARDS; DOSE COMMITMENTS;

<114> CONT.
CONTAINERS; SEA DISPOSAL; HAZARD ANALYSIS;
WASTES, HIGH-LEVEL; THEORETICAL STUDIES

<115>
Bealey, J.T., T.J. Headley, F.P. Glass, D.H. Strachan, and H.J. Kupfer, Sandia Laboratories, Albuquerque, NM; Rockwell Hanford Operations, Richland, WA.

Microstructural Characterization of Solidified Simulated Reactor Wasteforms.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 83-96), 563 pp. (1979)

Microstructural characterization of nuclear wasteforms for geologic disposal is essential to the understanding and evaluation of efficiency of the immobilization process and estimation of some physical properties such as leach rates and susceptibility to radiation damage. Techniques for characterizing such forms are scanning electron microscopy, transmission and scanning transmission electron microscopy, electron microprobe analysis, X-ray diffraction, X-ray fluorescence, quantitative microscopy, optical petrography, and helium pycnometry. Scanning electron microscopy is capable of discerning the devitrification processes in glass media. Transmission electron microscopy and scanning transmission electron microscopy are used to identify small and very small crystalline phases. Analyses from the electron microprobe provide quantitative compositions of the various phases and elemental distribution maps of both the glass and ceramic media. The presence and distribution of crystalline phases in the media can be determined by optical microscopy. Major crystalline phases may also be identified by X-ray diffraction analysis. Bulk composition is determined by X-ray fluorescence. Helium pycnometry is used to determine the volume percent of surface connected porosity present in ceramic media. A facility at Sandia houses the equipment required for these techniques. (RDV)

CHEMICAL PROPERTIES; PHYSICAL PROPERTIES;
METHODS; DIFFRACTION, X-RAY; CRYSTALLIZATION;
GLASS; CERAMICS; ELECTRON MICROPROBE;
MICROSCOPY; WASTES, SOLID; SOLIDIFICATION

<116>
Hoffman, T.L., Idaho National Engineering Laboratory, Idaho Falls, ID.

Corrosion Monitoring of Storage Bins for Radioactive Calcines.

ICP-1071; 18 pp. (1978, October)

High-level liquid wastes are generated at the Idaho Chemical Processing Plant as the result of uranium recovery operations from spent nuclear reactor fuel. These highly acidic wastes originate from the first cycle solvent extraction of the dissolved fuel elements and consist of the dissolved cladding elements, fission products and process adjustment chemicals. Management practice for high-level waste is to store the liquid waste safely in engineered bins of Types 405 and 304 (0.06 percent carbon maximum) stainless steels where it will be retrievable at all times. Corrosion evaluation of coupons withdrawn from these bins indicates excellent performance for the materials of construction of the bins. At exposure periods of up to six years the average penetration rates are 0.01 and 0.05 mils per year for Types 304 and 405 stainless steels, respectively. (Auth) (RAF)

CORROSION; MONITORING; WASTE STORAGE;
CONTAINERS; CALCINE; WASTES, HIGH-LEVEL; WASTES,
LIQUID; STEELS; SOLIDIFICATION; ALUMINUM;
ZIRCONIUM; ALLOYS; SITE SURVEILLANCE

<117>
Hoffman, T.L., Idaho National Engineering Laboratory, Idaho Falls, ID.

Corrosion Evaluation of Stainless Steels in Idaho Chemical Processing Plant High Level Radioactive Waste Service.

ICP-1072; 21 pp. (1976, June)

At the Idaho Chemical Processing Plant, several 300 series stainless steels were corrosion tested in raffinate storage tanks made of the same materials. After exposure to the raffinate solutions for periods up to nineteen years, maximum penetration rates of 0.1 mil per year were observed on Types 304L and 348 stainless steels and 0.2 mil per year on Type 316L stainless steel in aluminum-zirconium process first cycle raffinate, and only 0.006 mil per year on Types 304L and 348 stainless steels in second and third cycle raffinate. These favorable observations at significant exposure times give indication of a relatively long service life for the vessels. Continual measurements of liquid level in all tanks currently in use show that no leakage has occurred from any of the tanks due to tank failure from corrosion or other causes. (Auth) (RAF)

CORROSION; STEELS; WASTES, HIGH-LEVEL;
RAFFINATES; TANKS; WASTES, LIQUID; SITE
SURVEILLANCE; ALUMINUM; ZIRCONIUM; ALLOYS

<118>
Houser, C.A., L.S.T. Tsong, and R.B. White, The Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Characterization of Leached Surface Layers on Simulated High-Level Waste Glasses by Spotter-Induced Optical Emission.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 131-139), 563 pp. (1979)

The stability of glasses in leaching conditions is explored. In general the leaching of the alkalis causes the formation of an alkali-depleted layer on the outer surface which acts as a diffusion barrier to further atomic migration. For this study leaching was studied by measuring the

<118>

CONTAINER DESIGN AND PERFORMANCE

<118> CONT.

compositional profiles of H, Si, Al, the alkalis, the alkaline earths, and transition metals, such as Fe, Zr, and Mo. Preliminary results on a natural obsidian hydrated by the diffusion of ambient water show the alkali metals were clearly depleted at the surface. The same shape and depletion-depth for the alkalis suggest a common mechanism for their migration. For Mg and Ca the depth profiles were more irregular and exhibited less depletion than the alkalis. Of greatest interest was the apparent pile-up of both Ca and Mg on the near-surface layers, which indicates some reaction between Ca/or Mg and the alkali-depleted alumino-silicate framework possibly forming an alkaline earth silicate. The structure, however, of this layer has yet to be determined. The Si and Al profiles showed very little variation thus, suggesting that the alumino-silicate framework remains intact during leaching under neutral conditions. More work is planned. (NDV)

GLASS; LEACHING; ALKALINITY; RARE EARTHS; SILICATES; DIFFUSION; LATTICE; ALUMINUM; LABORATORY STUDIES

<119>

Jansen, G., and J.D. Keser, Battelle-Pacific Northwest Laboratories, Richland, WA.

Improved Heat Transfer from Radioactive Waste Canisters.

Transactions of the American Nuclear Society 17:327; CONF-731101; American Nuclear Society Winter Meeting, San Francisco, CA, November 11-15, 1973. (1973)

Canister design modifications to reduce the thermal resistance of the average path for heat flowing to the surroundings from stored solid radioactive wastes were studied at PNL. Modifications considered included (a) annular canisters, (b) internally finned canisters, (c) canisters with multiple internal cooling ports, (d) intimately mixed metal-waste composites, and (e) alternate layers of metal and heat-generating ceramic wastes. These were compared with simple cylindrical canisters containing only nonmetallic wastes. The limiting temperature was assumed to be the maximum waste temperature, and the volumetric heat generation rate was allowed to vary as the allowable total power varied. Parametric correlations for the improved internal heat transfer were developed for each of these systems. Cost comparisons between cylindrical, annular, and internally finned canisters for both stainless steel and carbon steel construction were based on estimates from architect engineers and vendors. Costs in dollars per unit volume were reduced to empirical equations involving the same variables as the heat transfer. Comparison of the two correlations gave correlations of the canister cost in dollars per watt of waste for each canister type. The cost advantage of annular over simple canisters passed through a maximum as size increased. Similar behavior was observed as a function of the number of internal fins, but at a much larger number of fins than is currently practicable. Neither cost advantage was greater than a factor of 2 over simple cylinders, and the resultant increase in operating costs for filling, shipping, and container integrity would tend to make their use marginally desirable. (Auth) (LKM)

CONTAINERS; DESIGN; HEAT RESISTANCE; HEAT TRANSFER; THERMAL PROPERTIES; WASTES, SOLID; WASTES, HIGH-LEVEL; WASTE DISPOSAL; WASTE STORAGE; COST BENEFIT ANALYSIS; EQUATIONS; THEORETICAL STUDIES

<120>

Jenkins, J.D., E.J. Allen, and E.D. Blakeman, Oak Ridge National Laboratory, Engineering Technology Division, Oak Ridge, TN.

Material Control and Accountability Procedures for a Waste Isolation Repository.

ORNL/TM-6162; 28 pp. (1978, May)

Addressed are control and accountability needs and requirements of a nuclear waste terminal storage facility. Such accountability will be to protect the health and welfare of the public by preventing theft, diversion, loss, and subsequent use of nuclear material by unauthorized personnel. Four types of waste are considered: high-level, cladding, intermediate-level, and low-level wastes. Standard containers will be used for storage and the high-level, cladding and intermediate-level wastes will have external shielding. Low-level wastes will probably be stored in 55 gallon drums. Lead, thorium, protactinium, uranium, neptunium, americium, cerium and all of their radioisotopes are considered. It is projected that by the year 2005, 35,320 canisters of HLW, CW, and ILW will be stored along with 76,000 drums of LLW. For accountability, record keeping is an absolute necessity. Other wastes may be accepted at the waste storage facilities if spent fuel reprocessing is not performed. Three levels of control are needed at the facility: identification and item control; tamper indication; and quantitative material assay. Present measures for high-level waste accountability are sufficient, which is to insure and demonstrate safe delivery of the material. The same is true for cladding wastes, due to the nature of the wastes. Intermediate-level wastes need both item identification and tamper indicators as a result of their nature. Low-level wastes are the most difficult to control because of the variety of sources, hence material control should occur at the place where they are encapsulated. Technical options for item control, tamper indication, and nuclear material assay are presented. (NDV)

WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, HIGH-LEVEL; CLADDING; CLASSIFICATION; CONTAINERS; HAZARD ANALYSIS; NUCLEAR MATERIALS; RADIONUCLIDES; SAFETY; REPOSITORY; WASTE VOLUME

<121>

Jenks, G.H., Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

Maximum Acceptable Temperatures of Wastes and Containers During Retrievable Geologic Storage.

Y/OWI/TM-42; 5 pp. (1977, August 17)

Estimates of maximum acceptable temperatures of waste and containers during retrievable

<121> CONT.

geologic storage were needed for use in evaluating and comparing conceptual designs for repositories for wastes from reprocessing and for spent fuel in several different rock types. Assumptions used in these estimates are that the length of the retrievability period is 5 years for all wastes except spent fuel, for which it is 25 years, and that all containers are of austenitic stainless steel, 304L, for high-level waste, and austenitic or carbon steel for the other wastes. Estimated maximum acceptable temperatures of waste and container during retrievable geologic storage (in deg. C) are as follows: for high-level waste in canisters - 375, as glass - 500, as calcine - 700, and as sintered glass ceramic - 800; for cladding and hardware in canisters - 375, as solid compressed waste - 200, and fused - 400; for intermediate-level TRU waste in canisters - 375, as glass - 500, in cement - 110; for spent fuel elements - 200; and for PuO_2 - 500. These figures are based on evidence in the literature relating to structural changes and stress cracking in austenitic stainless steel and carbon steel, transition and softening temperatures of borosilicate glasses, Cs volatility in calcine, reactivity of Zircaloy hulls during heating, etc. (Auth)(LKR)

TEMPERATURE; STORAGE, GEOLOGIC; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; CEMENTS; CONCRETES; GLASS; WASTES, TRANSURANIC; WASTES, SOLID; CONTAINERS; FUEL ELEMENTS

<122>

Johnstone, J.K., T.J. Headley, P.F. Hlava, and F.V. Stohl, Sandia Laboratories, Albuquerque, NM.

Characterization of a Titanate Based Ceramic for High Level Nuclear Waste Solidification.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 211-217), 563 pp. (1979)

To characterize a new titanate ceramic, several methods were employed to study its structure. Solidification occurs during an ion exchange process with commercial nuclear reactor wastes in a titanium-based inorganic ion exchange material and a synthetic zeolite. The titanate matrix, zeolite regions, metal particles and pores were discerned by light optical microscopy and electron microprobe techniques. X-ray diffraction techniques identified rutile as the dominant phase and a minor phase as the mixed rare earth titanate related to $Nd_2Ti_3O_{8.7}$. Other techniques showed the microstructure to be composed of a complex assemblage of small crystals superimposed upon a larger rutile matrix. Also present were large, amorphous silica regions. The fission product cations were concentrated in the fine-grained and to some extent the amorphous silica regions, with the rutile phase being pure. Identified phases included TiO_2 (rutile), $Cs_2O.Al_2O_3.4SiO_2$ (pollucite), Mo (metallic), $Gd_2Ti_2O_7$, Pd (metallic), and ZrO_2 (cubic and tetragonal). Other phases have yet to be identified. Thermal stability studies showed two distinct zones formed with the outer rim being crystallized and porous and the inner rim which is distinguishable from the interior. More studies are being made to fully understand the complexity of the thermal stability phenomenon. (EDV)

SOLIDIFICATION; CERAMICS; CRYSTALLIZATION; MINERALOGY; MICROSCOPY; TEMPERATURE; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; SILICA; TITANIUM COMPOUNDS; WASTES, RADIOACTIVE; FISSION PRODUCTS; WASTE TREATMENT; WASTE PROCESSING; LABORATORY STUDIES

<123>

Kernkraftanlagenbehörde, Stockholm, Sweden.

Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste. 8. Safety Analysis.

125 pp. (1978)

This report is a summarizing analysis of the handling, storage, and transportation of spent fuel and vitrified high-level waste in Sweden, and the long-term safety of the final storage of vitrified high-level waste in rock. It includes discussions of the following: general safety requirements including a brief background on laws, standards, and grounds for evaluation in relation to nuclear activities; a general survey of the various radionuclides and the generation of decay heat in spent fuel and vitrified high-level waste; the relative hazards of the various radionuclides through time; essential safety principals governing geologic storage of vitrified waste in Swedish bedrock; potentials for escape and dispersal of radioactivity; and long-term consequences of such dispersal in terms of radiation doses and medical effects. Quantities, transport, exposure pathways, and dose loads of radionuclides leached from glass are discussed on the basis of various conceivable dispersal mechanisms. It is assumed that the central storage facility for spent fuel will be in a rock cavern with approximately 30 m of rock cover. The fuel storage pools are designed as free-standing, thick-walled units of concrete, lined with stainless steel. The cooling system is designed to maintain the temperature at 25-30 C. A number of backup systems and a make-up water system can maintain cooling in the event of failure of the primary system. A dropped transport cask or fuel cassette has been found to give rise to radiation doses of less than 0.1 mrem in the environment, with a probability of 0.0008/yr. Normal activity releases during operation consist primarily of H 3 and Kr 85 (0.0001 mrem/yr.). Vitrified high-level waste will be handled and stored following principles established at the Marcoule plant in France. Analysis of transport accidents shows that probability of the most severe type -- ship collision with fire -- is about 3 x 10(E-6)/yr; release in such a circumstance is estimated at 6 rem for a maximally exposed individual, with a collective dose of 30,000 manrem if release occurs near a large city. High-level waste from reprocessing of spent fuel will be vitrified, encapsulated in Pb-Ti canisters, and embedded in buffer material (90% quartz, 10% bentonite) in rock 500 m deep. Consequences of the most severe breach of containment likely - that from a deep drinking water well nearby - is calculated to reach a maximum of 0.4 rem/30 days, attained after 200,000 years. The most probable dose load is less than 1% of the maximum. Regional and global dose levels under the worst circumstances would be 2000 manrem

<123> CONT.
over a 500 year period. Maximum calculated concentration increases for some radionuclides in a well and a lake near the final repository are as follows (in pCi/l): Ra 226 - 0.1 and 0.002, respectively; U - 30 and 0.6; Np 237 - 90 and 5; Cs 135 - 25 and 0.5. All calculations indicate values below the radiation protection limits for nuclear installations; the proposed method for final storage of high-level waste is therefore judged to be safe. Each chapter was separately input into the data base. (LKR)

Ra 226; U; Np 237; Cs 135

THEORETICAL STUDIES; SAFETY; HAZARD ANALYSIS; STORAGE, GEOLOGIC; WASTES, HIGH-LEVEL; SPENT FUELS; LEGISLATION; STANDARDS, INTERNATIONAL; RADIOACTIVE DECAY; GLASS; DOSE RATE; ENVIRONMENTAL EXPOSURE PATHWAY; DISPERSION; CONTAINERS; CONTAINMENT; COOLANTS; TRANSPORTATION; ACCIDENTS; TRITIUM; KRYPTON; ENCAPSULATION; WELLS; WASTES, SOLID; WASTE MANAGEMENT; WASTE DISPOSAL

<124>
Kenna, B.T., K.D. Murphy, and H.S. Levine, Sandia Laboratories, Albuquerque, NM.

Long-Term Elevated Temperature Leaching of Solid Waste Forms.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 157-160), 563 pp.(1979)

The leaching process of complete waste forms is not simple in nature and cannot be described by simple diffusion laws. The present work is an initial step toward understanding the basic leaching process(es). Both glass and ceramic waste forms containing 25 w/o simulated high-level waste have been undergoing Soxhlet leaching at elevated temperatures and are continuing. The leaching curves correspond to neither root nor linear time dependence, but there appears to be a periodic cycle of leaching. Immobile elements (Mo, Mn) exhibit maxima leaching about one month prior to mobile element (Na, Cs, Sr) maxima, but the minima of both coincide. The cyclic leaching is not due to inhomogeneous element distribution since electron microprobe results have assured elemental homogeneity. A surface "skin", depleted in mobile elements, forms on the glass samples and eventually loses integrity with the sample during the latter stages of the leaching cycle. A possible for cyclic leaching, a phenomenon which has been ignored in many past leaching studies, can be made on the basis of interaction between the elements themselves and with the matrix. (Auth) (Complete text)

LEACHING; DIFFUSION; GLASS; CERAMICS; WASTES, HIGH-LEVEL; LABORATORY STUDIES; TEMPERATURE; SPECTROMETRY

<125>
King, F.G., and W.H. Baker, Savannah River Laboratory, Aiken, SC.

Interim Storage of Spent Fuel Assemblies.

DP-MS-76-39; CONF-760701; Management of Waste from the LWR Fuel Cycle, Proceedings of an International Symposium, Denver, CO, July 11-16, 1976, (22 pp.).(1976)

Spent fuel discharged from light water reactors (LWR) is cooled at the reactor sites for at least 5 months to allow short-lived radioactive isotopes to decay. After this decay, fuel could be shipped to a chemical reprocessing plant for recovery of uranium and plutonium. Accordingly, storage facilities at LWR sites were originally intended for only short-term storage of spent fuel. More recently, spent fuel has been considered as a possible waste form suitable for interim storage or even ultimate disposal. Several alternatives have been demonstrated or proposed for retrievable storage of spent fuel for periods of up to 100 years. These include storage in water-cooled basins, air-cooled vaults, concrete surface silos, geologic formations, or near-surface heat sinks. The fuel cladding is usually considered the initial barrier for confinement of fuel core material for all storage concepts. The storage facility is designed to protect this cladding against mechanical, chemical, or thermal damage and to provide a safe, subcritical arrangement with adequate shielding under normal operating conditions or during extreme natural phenomena. Additional confinement is provided by a container if the fuel has been packaged or by cleanup of the coolant if bare fuel is stored. In all concepts, leaking fuel must be packaged in sealed containers to minimize radioactive releases. Water-cooled storage of spent fuel in near-surface cells of heavily reinforced concrete lined with stainless steel has been proven by about 30 years of operating experience at reactor sites and fuel reprocessing plants. Near-surface storage with forced-draft air cooling of HTGR (High Temperature Gas Reactor) fuels is being used by INEL (Idaho National Engineering Laboratory) and is feasible for unpackaged LWR fuel that has been out of the reactor at least 3 to 4 years. Natural-draft cooling of spent fuel has also been proposed, and demonstration programs are in progress for CANDU (Canada Deuterium Uranium) fuel. Spent fuel assemblies are sealed in thick low-carbon steel containers and placed in large cylindrical concrete housings (silos) located outdoors. The cooling is completely passive, requiring little maintenance and only minimal surveillance. Recent studies comparing the economics of interim storage of spent fuel (throwaway fuel cycle) to prompt chemical reprocessing conclude that disposal as fuel decreases the Nation's natural resources significantly and is not cost justified if both plutonium and uranium are recycled in the nuclear fuel cycle. (Auth)

SPENT FUELS; REACTORS, LIGHT WATER; RADIOACTIVE DECAY; FUEL REPROCESSING; URANIUM; PLUTONIUM; RETRIEVABILITY; WASTE STORAGE; WASTE DISPOSAL; STORAGE, GEOLOGIC; STORAGE, INTERIM; COOLANTS; WATER; AIR; CLADDING; SAFETY; SHIELDING; CONTAINERS; PACKAGING; CONCRETES; STEELS; REACTORS, HIGH-TEMPERATURE GAS-COOLED; REACTORS, CANDU; ECONOMICS; FUEL CYCLES

<126>
Lee, D.J., B.W. Veal, W. Chen, and G.S. Knapp, Argonne National Laboratory, Argonne, IL.

<126> CONT.
Bonding of Metal Oxides in Sodium Silicate Glass.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 97-107), 563 pp.(1979)

Fundamental information relating to the bonding properties of metal oxides in glasses is discussed. Two techniques were used to determine such information: X-ray photoemission spectroscopy (XPS) and extended X-ray absorption fine structure (EXAFS). For this study Fe_2O_3 and UO_2 in sodium silicate glass were employed. The combined XPS and EXAFS results show that the iron atom is surrounded by six oxygen atoms but is bonded to only three of them, and these oxygens are also bonded to the silicon atoms of the glass matrix. For UO_2 no evidence of bond breaking with subsequent reformation of uranium-oxygen bonds is found. Instead, the uranium oxides enter the silica lattice essentially as an additive without altering the bonding of the atoms with the glass matrix. To complete the study the coordination of the oxygen and surrounding uranium atoms will be determined. (NDW)

SOLIDIFICATION; ENCAPSULATION; URANIUM; IRON COMPOUNDS; LATTICE; GLASS; SILICATES; OXYGEN; LABORATORY STUDIES

<127>
LaRiviere, E.C., D.C. Nelson, and W.A. Zweifel, Atlantic Richfield Hanford Company, Richland, WA.

Development of National Waste Terminal Storage Program Waste Packaging Criteria.

Transactions of the American Nuclear Society 26:287-288; CONF-770611; American Nuclear Society Annual Meeting, New York, NY, June 12-16, 1977, (pp. 287-288).(1977)

Waste containerization studies are being conducted by Atlantic Richfield Hanford Company as part of the systems studies of the NWTS Program to develop waste acceptance criteria for commercial nuclear waste packages. The studies will attempt to optimize canister design for material, size, heat content, configuration, etc. Wastes being considered are high-level and intermediate-level, intended for eventual deep geologic storage. Intermediate-level waste is defined as TRU contaminated waste with a lower limit of 10 mCi/g and an upper limit at the bottom of the HLW range; wastes expected to be included in this category are cladding hulls, gases, failed equipment, and decontamination and decommissioning wastes. Gases would include Kr 85, I 129, C 14, and H 3. Kr would likely be packaged under pressure, and I as a solid in metal containers. Development of separation and packaging techniques for C 14 and H 3 is in the early stages. Failed equipment and wastes from decontamination and decommissioning of nuclear facilities would likely be packaged in standard-size containers. For the "throw-away" fuel cycle case, storage of spent PWR and BWR fuels will be considered. Allied General Nuclear Services has proposed the use of a 48-in. diameter x 8-ft long container for metal cladding hulls in their reprocessing plant. Current status of design criteria for solidifed high-level waste containers is reflected in the reference canister proposed by Sandia Laboratories for the WIPP; the canister is a double-shell right circular cylinder, 16 in. o.d. by 10 ft long. Thermal power is 3.5 kW. The inner container is carbon steel. Preliminary design criteria for HLW canisters will be issued in 1977; those for ILW packages will be issued in early 1978. As new data are developed, these criteria will be revised and reissued; additional design verification tests will be conducted as necessary. (Auth)(LKM)

CONTAINERS; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, TRANSURANIC; DESIGN; STORAGE, GEOLOGIC; WASTE STORAGE; CLADDING; GASES; EQUIPMENT; KRYPTON; IODINE; TRITIUM; CARBON; WASTES, SOLID; CANISTERS; REGULATIONS, FEDERAL

<128>
Laede, F.

Glass as the First Barrier for Long-Term Storage of Highly Active Radioactive Wastes.

CB-35802; OECD Nuclear Energy Agency and the Commission of the European Communities, Proceedings of a Workshop, Ispra, Italy, May 23-27, 1977, (pp. 50-77).(1977, October)

Referring to studies on the properties of glasses, the report discusses the characteristics of glasses used in the long-term fixation of radioactive elements contained in fission-product solutions. An estimation of the time necessary for confinement of the containers, and a review of the studies on behavior of glasses made by the Commission of the European Communities (CEA) at Marcoule, is presented. For fission products alone, the necessary confinement time, and therefore the durability of the silica-borate glass barrier, is no longer than 1000 years. The glasses studied have stability properties sufficient to be a primary barrier during this time. For the actinides, confinement should continue beyond 100 years, but not exceed 10,000 years. Stability tests on glasses containing beta or gamma emitters were conducted by simulating self-irradiation with doses of 1.2 x 10(E+11) rad. No measurable energy accumulation or structural modifications appeared and leaching-out rates were not affected. Thermal tests conducted on these glasses at 500 and 600 deg. C for one year indicated that silico-borate glasses with silica content above 80% generally behaved well. High contents of fission-product oxides (25%) and the presence of molybdenum favor crystallization, whereas the presence of boron is beneficial. Leaching-out tests were carried out on glasses with little tendency to crystallize. Rates varied relatively with composition and were between 10(E-6) and 10(E-9) g per sq cm per day. Diffusion coefficients were 10(E-19) to 10(E-6)sq cm per day for strontium and cesium, and between 10(E-21) and 10(E-20) sq cm per day for ruthenium and cerium. In glasses containing alpha emitters, ag'-g was simulated by doping glass blocks with Pu 238, Am 241, and Cm 244. Measurements of helium occluded in the glass were 5.2 cu mm and 5 cu mm per g of glass for Pu 238 and Am 241 respectively, with results from the Cm 244 not available yet. Mean values for leaching out rates were 2 x

<128> CONT.
10(E-7) for Pu 238, and 5 x 10(E-9) for Am 241. Thermal shock tests showed that the glass blocks will be fissured and their surface area will then be multiplied by a factor of 10. Storage in concreted trenches at the surface seems to be needed for cooling during the first years to get over the thermal problems. As soon as cooling by natural convection is sufficient, this type of storage will be advantageous in that it will require very little maintenance and it will offer possibilities of monitoring and taking corrective action. Subsequently, since periods of 1000 years and more will be required, storage in geological formations will necessarily be irreversible, and will then have to constitute the indispensible complementary barrier. (CAJ)

Several graphs and tables are included.

Leaching Rate; Diffusion Coefficient

Pu 239; Am 241; Cs 244

CONTAINERS; CONTAINER INTEGRITY; GLASS; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE; WASTE STORAGE; VITRIFICATION; FIELD STUDIES; THERMAL PROPERTIES; LEACHING; SILICA; BORON; MOLYBDENUM; FISSION PRODUCTS; FISSURES; LABORATORY STUDIES; WASTE DISPOSAL; STORAGE, GEOLOGIC; DIFFUSION; ALPHA PARTICLES; BETA PARTICLES

<129>
Lini, D.C., Atlantic Richfield Hanford Company, Richland, WA.

Compilation of Hanford Corrosion Studies.

ARH-ST-111; 36 pp.(1975, July)

This report reviews experimental corrosion studies conducted at the Hanford Site over the last 30 years, identifying types and rates of corrosion, describing techniques used in gathering corrosion data, and indicating how these data contribute to tank failure predictions. Studies have included general and pitting corrosion, stress corrosion cracking, corrosivity of solidified high-level waste, and cathodic protection. Early laboratory studies were conducted using coupon immersion tests. Results indicated that the rate of general corrosion is acceptable for storage of caustic-nitrate solutions in mild steel containers. Pitting corrosion data have not been useful in predictions. A few in-tank studies were made using test coupons exposed to actual wastes; results tended to confirm the bulk of the laboratory experiments [rate of general corrosion approximately 5 X 10(E-5) in./mo in both cases]. In the area of stress corrosion cracking (SCC), Hanesz and Moore showed that the 6-8 M nitrate concentrations found in high-level liquid waste cause cracking in mild steel liners at temperatures above 100 C. SCC can also be caused by residual stress in weld areas. The corrosive effects of simulated salt cake were studied. Results showed corrosion rates from 0.2-2.2 mils/yr. Long-term immersion studies are ongoing. Cracking prevention methods have been demonstrated. Thermal-stress annealing has been shown highly effective and is now being applied to all double-walled storage tanks; cathodic protection is presently being investigated. (LKN)

Corrosion Rate

CORROSION; TANKS; NITROGEN COMPOUNDS; CRACKING; WASTES, HIGH-LEVEL; STRESS; WASTES, SOLID; WASTES, LIQUID; STRESS; ELECTROCHEMISTRY; NITRATES; STEELS; CONTAINERS; LABORATORY STUDIES; FIELD STUDIES; SALTS; EVALUATION; PREDICTIONS; CORROSION RATE

<130>
MacMillian, W.W., and R. Rey, The Pennsylvania State University, Material: Research Laboratory, University Park, PA.

Corrosion of Metals in a Hot, High Pressure Brine Environment.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 29-December 1, 1978, (pp. 69-89), 77 pp. (1979)

Candidate containers for radioactive waste solids will be exposed to corrosion from both inside (by interaction with the waste form) and the outside (by interaction with the host rock and probable aqueous solution). The most corrosive of the likely environments are the brines at the temperatures on the container surface (which we limit in this study to about 300 C). This paper will present the results of a preliminary investigation of the relative corrosion resistances of seven metals (Ta, Hf, Au, Ag, Pb, Ti and Cu) and eight alloys (1018 mild steel, 316 stainless steel, Everder silicon bronze, Monel, Coronze, Hastalloy C, Zircaloy-2 and a-brass) in a concentrated brine at temperatures and pressures of up to 250 C and 15000 psi, respectively, simulating early repository conditions. The chosen brine, which contains 3.38 gm $MgSO_4$, 15.80 gm Na_2SO_4 and 342.52 gm NaCl per liter, simulates the groundwater at the Waste Isolation Pilot Plant in Carlsbad, NM. In order to avoid the galvanic effects that would result from encapsulation in Au tubes, the present experiments involve enclosing the metal foil specimens in brine-filled sealed Teflon containers and heating them in a water-filled autoclave at the desired temperature and pressure for times ranging from a few days to a few weeks. The nature and the extent of the resultant corrosion has been determined in the first instance from weight change measurements and X-ray studies of the corrosion products formed on the foils. No catastrophic breakdown has been found with any of the above metals. (Auth)

Abstract is complete text. The paper was not included in the Proceedings.

WASTES, SOLID; CORROSION; CONTAINERS; CONTAINER INTEGRITY; METALS; FOILS; TEMPERATURE; LABORATORY STUDIES

<131>
Malow, G., and H. Andresen, Hahn-Meitner-Institut fur Kernforschung Berlin, GmbH, Berlin, German Federal Republic.

Helium Formation from Alpha-Decay and Its Significance for Radioactive Waste Glasses.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al.,

<131> CONT.
Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 109-115), 563 pp. (1979)

Helium behavior in irradiated borosilicate glasses containing actinides was investigated. In the temperature range of 456-600 K He diffusion was found to be dependent on the radiation dose, and the activation enthalpy rises up to about 85 kJ/mole. As the majority of the waste constituents are incorporated into the glass structure as network modifying ions, a high waste content in the glasses should inhibit He diffusion. Crystal growth in glasses containing radwastes will be diffusion controlled. Several growth rates for several crystal phases were estimated in borosilicate glass. From diffusion measurements and calculations the desirable host phases for fission products such as pollucite and fluorite or scheelite structures may not form at the relatively low storage temperatures. However, Na and Li containing species such as metasilicates may be more readily formed due to their higher mobility and low activation energies. This fact is important for the development of glass ceramics. (NDV)

SOLIDIFICATION; ENCAPSULATION; GLASS; HELIUM; SILICATES; DIFFUSION; LATTICE; CERAMICS; TEMPERATURE; THERMAL PROPERTIES; ACTINIDES; RCRCR COMPOUNDS; LABORATORY STUDIES

<132>
Rattsson, E., Swedish Corrosion Institute, Stockholm, Sweden.

Corrosion Resistance of Canisters for Final Disposal of Spent Nuclear Fuel.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 271-281), 563 pp. (1979)

Swedish scientists evaluated aspects of three canister types proposed for final disposal of nuclear reactor waste in 500 m deep boreholes in low-permeability igneous rock. Under storage conditions the canisters would be embedded one per borehole in bentonite or quartz sand/bentonite, and the connecting horizontal tunnels would be filled with the sand/bentonite mixture which would become saturated with ground water and swell to form a low-permeability buffer. The first canister type studied was titanium with a wall thickness of 6 mm and a 100 mm thick lining of oxygen-free lead, to hold reprocessed and vitrified waste. Titanium has good corrosion resistance in water solutions over a wide range of pH values because it forms a layer of titanium dioxide. Corrosion in chloride solutions is in the range of 0.1 - 0.25 um/yr. The general conclusion is that corrosion is of little threat, though if pH and temperature are unsuitable, pitting and crevice corrosion are possible with warm chloride solutions. Also, hydrogen embrittlement of the titanium could occur if the solubility is by anodic dissolution, but perforation would take thousands of years. Service life of the canister is estimated to be thousands to tens of thousands of years. The second alternative studied was copper canisters for spent unreprocessed fuel, in which rods would be enclosed in oxygen-free high conductivity copper with 200 mm thick walls. Surrounding the canister in the borehole would be bentonite with a sulfide content no greater than 200 mg/kg, with low permeability [10(E+12) m/s] and a high swelling capacity when moistened. The sand/bentonite in the tunnels would be given an admixture of iron phosphate to remove oxygen. Copper solubility in oxygen-free pure water is low, but copper can be attacked by dissolved oxygen in water at a high redox potential. Copper oxidation by sulfate or nitrate is possible, but slow in the absence of bacteria. Estimated service life of this canister is hundreds of thousands of years. The third canister alternative studied was alumina for spent unreprocessed fuel in which the rods are coiled up and enclosed in 99.8% Al2O3 with a wall thickness of 100 mm. Corrosion tests with total immersion for half a year in pure water of pH 7.0 - 9.3 at 90 C show corrosion rates of 0.01 - 0.07 um/yr, but a lower rate might be expected in storage where a protective surface coating may form. Delayed fracture could possibly result from growth of surface defects under high mechanical tension. Service life of the canister cannot properly be estimated without further study, but results indicate a corrosion rate of less than 0.1 um/yr. (DCU)

Corrosion Rate

WASTES, RADIOACTIVE; WASTE STORAGE; STORAGE, GEOLOGIC; WASTE DISPOSAL; SPENT FUELS; IGNEOUS ROCKS; CONTAINERS; CANISTERS; CHEMISTRY; CHEMICAL PROPERTIES; CORROSION; METALS; LEAD; TITANIUM; COPPER; ALUMINUM COMPOUNDS; BENTONITES; SANDS; PERMEABILITY; pH; SOLUBILITY; BOREHOLES

<133>
McDonell, W.R., and S. Dillich, Savannah River Laboratory, Aiken, SC.

Effects of Radiations from Transuranium Nuclides on Container Surfaces.

DP-MS-78-9. (1978)

Container materials for short-lived transuranium nuclides such as Cm 242 and Cf 252 may be implanted with sufficiently high concentrations of helium to cause surface blistering and other deleterious effects in some environments. Platinum and stainless steel surfaces adjacent to solid Cm 242-oxide deposits showed no blisters after room-temperature exposures, but formed porous, distorted layers on postexposure heating. Platinum surfaces adjacent to solid Cf 252-oxide deposits, in contrast, developed well-defined blisters with a variety of configurations during room-temperature exposures. Formation of the blisters at low temperatures was attributed to the fission-fragment-induced agglomeration of implanted helium, coupled with a stress-relieving transport of atoms displaced by fission fragments into the blister walls. High-temperature exposures produced less well-defined globular protrusions at such lower concentrations of implanted helium. Immersion of glass or platinum specimens in dilute Cm 242 or Cf 252 solutions caused no significant surface damage, and on subsequent reaction with acidic or caustic reagents other than HF, the surfaces displayed no well-defined effect of the radiation exposures. HF solutions preferentially

<133>

CONTAINER DESIGN AND PERFORMANCE

<133> CONT.
etched fission tracks in Cf 252-exposed glass specimens. No significant retention of Cf 252 implanted in container surfaces by fission fragment impact was detected. (Auth)

Photographs are given of container surfaces exposed to Cf 252 and Cm 242 solutions at different temperatures and for reacted with acids or bases.

Cf 252; Cm 242

RADIATION EFFECTS; WASTES, TRANSURANIC; WASTES, SOLID; WASTES, LIQUID; CURIUM 242; CALIFORNIUM 252; OXIDES; ALPHA PARTICLES; HELIUM; CONTAINERS; CONTAINER INTEGRITY; SURFACE PROPERTIES; PLATINUM; STAINLESS STEELS; THERMAL PROPERTIES; GLASS; ACIDS; FISSION

<134>
Benlowe, H.O., G. Bockelman, D.L. Close, and L.G. Speir, Los Alamos Scientific Laboratory, Los Alamos, NM.

Neutron Interrogator Assay System for the Idaho Chemical Processing Plant Waste Canisters and Spent Fuels Preliminary Description and Operating Procedures Manual.

LA-7250-M; 36 pp. (1978, May)

In striving for improved nuclear material control and accounting, an accountancy problem was identified in the waste canisters from the Idaho Chemical Processing Plant (ICPP) fluorinel dissolver. The sludge is nonuniform and contains large quantities of zirconium, fission products, and thermal neutron poisons such as boron and cadmium. No proven method or instrumentation exists for measuring this type of material. A neutron interrogation assay system is being designed for the measurement of waste canisters and spent fuel packages. The assay system is a Cf 252 "Shuffler" that employs a cyclic sequence of fast-neutron interrogation with a Cf 252 source followed by delayed-neutron counting to determine the U 235 content. (Auth) (RAP)

WASTES, RADIOACTIVE; NEUTRONS; CANISTERS; SPENT FUELS; PACKAGING; URANIUM 235; EQUIPMENT; POISONS; CADMIUM; BORON; MEASUREMENTS; MOISTURE; CALIFORNIUM 252; CURIUM 242; CURIUM 244; BACKGROUND RADIATION; SHIELDING; RADIATION SOURCES; TRITIUM; CALCULATIONS

<135>
Henon, H.B., Hanford Engineering Development Laboratory, Richland, WA.

Spent Fuel Handling and Packaging Program. A Survey of Hot Cell Facilities.

HEDL-TME-78-53; 21 pp. (1978, July)

Hot cell facilities in the United States were surveyed to determine their capabilities for conducting integral fuel assembly and individual fuel rod examinations that are required in support of the Spent Fuel Handling and Packaging Program. The ability to receive, handle, disassemble and reconstitute full-length light water reactor spent fuel assemblies, and the ability to conduct nondestructive and destructive examinations on full-length fuel rods were of particular interest. The hot cell facilities in this survey are the Engineer Maintenance and Disassembly (EMAD) Facility in Nevada; Los Alamos Scientific Laboratory (LASL) at Los Alamos; Vallecitos Nuclear Center (VNC) near Pleasanton, California; Hanford Engineering Development Laboratory (HEDL) at Richland, Washington; Battelle Columbus Laboratories (BCL) Hot Cell Facility near Columbus, Ohio; and the Lynchburg Research Center (LRC) at Lynchburg, Virginia. All of the facilities surveyed, except EMAD, were found to have the capability for conducting destructive examinations. However, only BCL, LRC and VNC have past experience and are currently involved in examining commercial LWR fuel rods. Among these three, only BCL has had experience in examining both pressurized and boiling water reactors. (RAP)

SPENT FUELS; PACKAGING; HOT CELLS; WASTE STORAGE; REACTORS, PRESSURIZED WATER; REACTORS, BOILING WATER; REACTORS, LIGHT WATER; PREPARATION; SURVEYS

<136>
Berlini, R.J., D.L. Cash, and K. Terada, Rockwell International, Atomics International Division, Rocky Flats Plant, Golden, CO.

Radioactive Waste Package Development at the Rocky Flats Plant.

CONF-760813; IAEA-SM-10/26; Design Construction and Testing of Packaging for the Safe Transport of Radioactive Materials, Proceedings of a Symposium, Vienna, Austria, August 23-27, 1976. International Atomic Energy Agency, Vienna, (13 pp.). (1976)

Information is presented on some of the historical and current package developments for the plutonium 239 wastes generated at the Rocky Flats Plant. The two basic packages used for waste containment during transportation and storage are the steel drum and the plywood box. For steel drums, a discussion is presented on a series of liners and linear characteristics designed to provide long package life. This discussion includes data on the liner specifications, materials of construction, sealing techniques, seal strengths, and drop tests. For the plywood box, some experiences and developments are described on the design, construction, and drop testing of boxes coated with fiberglass reinforced plastic (FRP). Cost data and comparisons as appropriate for different drum liners and for the FRP coated plywood box are also included. (Auth) (RAP)

Drawings are given of drum liner components and flange sealing techniques.

WASTES, TRANSURANIC; WASTES, SOLID; PLUTONIUM 239; PACKAGING; TRANSPORTATION; DRUMS; CONTAINERS; STEELS; RESINS; WOOD; LINERS; SEAL MATERIALS; DESIGN; PLASTICS; FOUNDERS; COST BENEFIT ANALYSIS; BOXES

<137>
Molecke, M.A., Sandia Laboratories, Albuquerque, NM.

Waste Isolation Pilot Plant In Situ Experimental

<137> CONT.
Program for High-Level Waste.

T/ORI/SUB-77/10260; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 55-569), 92 pp. (1977, August 20)

In situ high-level waste experiments are planned at the WIPP along with inter-related laboratory and bench-scale studies. These in situ experiments will begin in approximately 1985, depending on the availability of commercially reprocessed waste plus delays in the construction schedule for the WIPP. One set of experiments involves emplacing bare HLW fragments into direct contact with the bedded salt environment. A second set utilizes full-size canisters of waste emplaced in the salt in the same manner as is planned for a future HLW repository. The bare waste experiments will study in an accelerated manner waste-salt-brine interactions including matrix integrity/degradation, brine leaching, system chemistry, and potential radionuclide migration through the salt bed. Using full-size canisters of HLW in situ would permit demonstration of operational effectiveness and safety. Experiments will evaluate corrosion and compatibility interactions between the waste matrix, canister, and overpack materials, getter materials, stored energy, waste inventory, etc. Using full-sized canisters also allows demonstration of engineered retrievability of wastes, if necessary, at the end of experimentation. (Auth) (LKN)

Only the abstract is given.

WIPP; WASTES, HIGH-LEVEL; SALT DEPOSITS; WASTES, SOLID; CONTAINERS; REPOSITORY; STORAGE, GEOLOGIC; BRINES; LEACHING; GEOCHEMISTRY; RADIONUCLIDE MIGRATION; CORROSION; RETRIEVABILITY; FIELD STUDIES; SITE SELECTION; WASTE-ROCK INTERACTIONS

<138>
Gdej, E., E. Hilgert, and H. Gerads, Kernforschungsanlage Julich GmbH, Institut for Chemical Technology, Julich, Germany.

Investigations of the Volatility of Cesium from Aluminosilicates by Mass Spectrometry.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G. J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 227-229), 563 pp. (1979)

High-temperature mass-spectrometric studies on volatility were carried out to investigate thermodynamic properties and the influence of glass forming additives. Samples used belong to the ternary system Cs2O-Al2O3-SiO2. Some samples contained lower content of cesium so their phase composition could be studied. The crystalline compounds have the stoichiometric compositions $CsAlSiO_4$, $CsAlSi_2O_6$, $CsAlSi_5O_{12}$. From vapor pressure data it is possible to determine the phase or the composition in which the cesium is retained in the ceramic or glass by the Horts-Knudsen equation. (ERF)

SPECTROMETRY, MASS; CHEMICAL COMPOSITION; THERMAL PROPERTIES; EQUATIONS; GLASS; CRYSTALLIZATION; LABORATORY STUDIES

<139>
Gudrejcin, E.S., Savannah River Laboratory, Aiken, SC.

Investigation of Cooling Coil Corrosion in Radioactive-Waste Storage Tanks.

DP-1425; 21 pp. (1977, January)

Pitting of cooling coils in high-heat waste storage tanks after sludge removal operations was investigated to determine the cause and suggest preventative measures. Tests were conducted using airlatex cooling coils of A 106 steel immersed in various test solutions. Heated synthetic sludge supernate and synthetic sludge produced only mild pitting (less than 0.025 mm/month); a ten thousandfold dilution of supernate increased pitting by a factor of 2, to 0.076 mm/month - still too low to account for observed pitting in full-scale coils. Increased sulfate in the sludge, combined with dilution of basic nitrite inhibitor in the waste could result in rapid pitting; polarization studies showed that in dilute, 0.03 M sulfate, 10(E-4)-10(E-3) M nitrite, simulated waste supernate, basic nitrite is not effective as an anodic inhibitor, and predicted corrosion rates (1.27 mm/month), if localized as pitting, could be as high as the observed rate. Attack is probably initiated at oxygen depletion cells. The sulfate, which was leached from the sludge, produces a conductive solution that could generate high current densities at the corroding steel surface. Recommendations for prevention of pitting during future sludge removal operations are that the sludge be slurried using waste diluted no more than one hundredfold with water or using a 500 ppm nitrite-water solution at pH 12. (LKN)

CORROSION; SLUDGES; TANKS; WASTE STORAGE; NITRITES; SULFUR COMPOUNDS; WASTE TREATMENT; WASTES, RADIOACTIVE; LABORATORY STUDIES; WASTES, HIGH-LEVEL

<140>
Parimoau, W.G., RE/SPEC, Inc., Rapid City, SD.

Thermoelastic/Plastic Analysis of Waste-Container Sleeve: 8. Air Gap Influence on Hole Closure.

ORNL/SUB-8269/3; 6 pp. (1975, March 21)

This report presents the results of an axially symmetric thermoelastic/plastic finite element analysis of the closure of an air-gap between the borehole wall and a waste-container/sleeve assembly positioned in the lower 10 ft of an 18 ft deep borehole centrally located in a typical 18 ft wide X 17 ft high New Mexico bedded salt repository pilot-plant room adjacent to a 60 ft wide pillar. The objective of the analysis is to obtain an estimate of the maximum air-gap closure expected over a 10 year heating period. In order to simulate the air-gap influence on borehole closure, the SALT/87 Model was used, with the exception that the elements representing the waste can were replaced by "air elements" [Young's Modulus of 10(E-4) psi] that experienced the same

<180>

CONTAINER DESIGN AND PERFORMANCE

<181> CONT.
temperatures as the waste can but offered no resistance to borehole closure. The air-gap analysis shows that the borehole wall moves inward at most 0.081 in at the hole collar and 0.040 in the lower 10 ft of the hole where the waste container/sleeve would be positioned. Outward thermal expansion of the waste/container/sleeve would be less, so that total air-gap closure would be on the order of hundredths of an inch at most. The width of an acceptable air-gap does not appear to depend on the thermomechanical behavior of the waste-container/sleeve and salt. A borehole sufficiently oversize with respect to sleeve diameter to allow emplacement speed and convenience but not so large as to represent unnecessary drilling machine capability would probably provide an acceptable air-gap. Such an air-gap would eliminate the engineering necessity for the sleeve. It would seem worthwhile to explore this possibility further by using a refined finite element mesh to obtain more detailed data at the borehole wall and by experimental tests in the laboratory. (AWB) (LKH)

BOREHOLES; MODELS; MODELS, MATHEMATICAL; COMPUTER PROGRAMS; COMPUTER CODES; SALT DEPOSITS; STORAGE, GEOLOGIC; WASTE STORAGE; DEPOSITORY; CONTAINERS; THERMAL PROPERTIES; ELASTOPLASTICITY; ROCK MECHANICS; THEORETICAL STUDIES

<182>
Pinchback, T.R., and D.A. Knecht, EG and G Idaho Incorporated, Idaho Falls, ID; Allied Chemical Corporation, Idaho Falls, ID.

The Development of Process and Storage Materials Suitable for Krypton 85 Waste Management.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 879-885), 563 pp. (1979)

Federal regulations require that no more than 10% of the krypton 85 produced in nuclear fuel after 1983 be released to the atmosphere. Therefore, techniques for its removal and storage must be developed. It may be stored as a pressurized gas or as a solid after immobilization in zeolite. The decay product of krypton 85, rubidium, along with other gas impurities, is a potentially corrosive agent to containers, valves, and process equipment. A test program has been in progress for evaluating the compatibility of candidate materials with liquid rubidium and rubidium compounds. Stressed "C" rings were selected for initial screening exposure in liquid rubidium, since this test provides a simple, low-cost method of determining corrosion and embrittlement problems for the candidate materials. Tested materials included: 4130 steel; 304, 316, and 347 stainless steel; Nitronic 50; A286; Monel 400; and Inconel 600. Screening tests for these materials immersed in high purity liquid rubidium showed no liquid embrittlement for statistically stressed, smooth "C" rings at temperatures between 400 and 672 K. Potentially injurious localized corrosion in the form of pitting was observed for 304 stainless steel at 672 K. All other materials showed good performance in the temperature range proposed for krypton 85 gas cylinder storage. Type 304 stainless steel showed significant general corrosion and intergranular attack in liquid rubidium of lower purity between 793 and 893 K. Type 304 stainless steel is not recommended for hardware which may encounter similar service conditions. (AWB) (DCW)

WASTES, RADIOACTIVE; WASTE STORAGE; WASTE DISPOSAL; WASTES, GASEOUS; WASTES, SOLID; KRYPTON 85; ZEOLITES; IMMOBILIZATION; CONTAINERS; CORROSION; LABORATORY STUDIES; REGULATIONS, FEDERAL; RUBIDIUM; RELEASE LIMITS

<183>
Port, E.A., Northwestern University, Center for Radiation Safety, Chicago, IL.

An Improved Receptacle for Radioactive Waste.

Health Physics 29(5):801-802. (1975, November)

At Northwestern University liquid waste is collected in non-breakable polyethylene 5-gal jugs, and solid waste is collected in step-on waste cans lined with disposable plastic bags. Because these disposable plastic bags present a number of problems, an alternative for handling dry waste was sought. At the present time all radioactive waste is placed in fiber drums with heavy polyethylene liners. The solution finally adopted to minimize breakage of the bags involved the use of a funnel top which is available as a Rubbermaid part No. 3540, and a fiberboard drum Grief Bros. No. V22-S25-923 made to the same dimensions as the standard waste receptacles. The drum has a polyethylene liner which is 18 in. longer than the drum. The liner is pulled up through the funnel top openings and drawn down over the funnel top and upper portion of the drum, protecting the funnel top and upper portion of the drum from contamination. When the drum is filled the inverted liner is lifted straight up and tied off at the mouth of the funnel top. The funnel top is removed and the drum head is installed on the drum in the laboratory. At the time of the removal of the full drum from the laboratory, a new drum is placed in the laboratory and the funnel top fitted. The anticipated reduction in waste costs should result from both the improved utilization of space and the reduction of waste pickups required. (JMT)

CONTAINERS; WASTE DISPOSAL; WASTES, RADIOACTIVE; WASTES, LOW-LEVEL; DRUMS; LINERS

<184>
Rack, H.J., and G.A. Knorovsky, Sandia Laboratories, Albuquerque, NM.

An Assessment of Stress-Strain Data Suitable for Finite-Element Elastic-Plastic Analysis of Shipping Containers.

NUREG/CR-0091; SAND-77-1872; 69 pp. (1978, September)

This report is a review of stress-strain data available in the literature which describe the influence of strain rate and temperature on the mechanical response of materials currently being used for LWR spent fuel shipping containers. Selection of data has been limited to that which is suitable for use in finite-element elastic-plastic

<183> CONT.
analysis of shipping containers (e.g., they must include complete material history profiles). This survey considers only uniaxial deformation, and excludes information on fracture, creep, and cyclic loading conditions. Materials used in typical shipping cask designs are listed, along with their procurement specifications; available mechanical properties data, particularly stress-strain curves, are discussed. Based on this information the following recommendations are made for further work which will be required to complete the data base: define the temperature, strain-rate, and strain regions for which strain-induced martensite and dynamic strain-aging will control the stress-strain behavior of austenitic stainless steels used in LWR shipping casks; establish a data base for selected austenitic stainless steel, "chemical" lead, and as-cast alpha-uranium, including the influence of strain-rate and temperature on tensile, compressive, and shear properties, the influence of chemistry variation on mechanical properties, and the influence of residual stress level and test environment on the mechanical behavior of as-cast alpha-uranium; develop constitutive equations describing stress-strain behavior of cask materials under normal and abnormal modes of deformation; establish the thermal expansion behavior of 216, 317, 321, and 347 stainless steel between -40 and 320 C; establish the thermal expansion behavior of typical product forms of alpha-uranium used in shipping casks, particularly the expected anisotropic orientation dependence of thermal expansivity; extend Young's and shear moduli measurements for austenitic stainless steels to -40 C; determine the elastic properties of 216, 308, 317, and 347 stainless steel; and determine the elastic properties of selected dilute uranium alloys (e.g., U-2Mo). The primary emphasis of all of these studies should be a systematic and quantitative assessment including pertinent microstructural information rather than the largely qualitative information available at present. (Auth) (LKH)

CONTAINERS; CASKS; PHYSICAL PROPERTIES; THERMAL PROPERTIES; STRESS; STRAIN; SPENT FUELS; TRANSPORTATION; ELASTOPLASTICITY; DEFORMATION; STAINLESS STEELS; LEAD; ALLOYS; URANIUM; EQUATIONS; REVIEWS; DESIGN; REACTORS, LIGHT WATER; WASTES, RADIOACTIVE; WASTE MANAGEMENT

<184>
Rhodes, D.W., Idaho National Engineering Laboratory, Idaho Falls, ID.

Primary Source of Radionuclides Leaking into the Idaho Chemical Processing Plant Fuel Storage Basin Water.

ICP-1112; 12 pp.(1977, February)

A study was made to define the type of fuel that is the primary source of radionuclides "leaking" into the fuel storage basin at the Idaho Chemical Processing Plant. The isotopic compositions of plutonium and uranium in sludge on the basin floor, in the fuel storage basin water, and in the various types of fuel, as indicated by analysis of process solutions, was used to determine that the stainless steel-type (EBR-II) fuel was the main "leaking" fuel. A comparison of the Sr 90/Sr 89 ratio in the basin water with calculated values for fuel indicated that the leaking fuel is less than one year cooled. (Auth) (RAF)

PLUTONIUM 239; URANIUM 235; URANIUM 236; FUELS; SPENT FUELS; STORAGE, ABOVEGROUND; SLUDGES; COOLANTS; WATER; FISSION PRODUCTS; ION EXCHANGE; ISOTOPE RATIOS; STRONTIUM 89; STRONTIUM 90; WASTE STORAGE; LANTHANUM 140; BARIUM 140; IODINE 131; LEAKS

<185>
Ringwood, A.E., Australian National University, Research School of Earth Sciences, Canberra, Australia Capital Territory, Australia.

Incorporation of High-Level Radwaste in SYNROC.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978, (p. 21), 77 pp.(1978)

SYNROC is a synthetic rock composed of some or all of the phases: hollandite ($BaAl_2Ti_6O_{16}$), cubic zirconia (ZrO_2-Ca,Ti ss.) perovskite ($CaTiO_3$), celsian ($BaAl_2Si_2O_8$), "kalsilite" ($KAlSiO_4$) and leucite ($KAlSi_2O_6$ ss.). The mineral assemblage may be formed by direct crystallization from a melt cooled below 1300 C or by hot pressing under subsolidus conditions at 1100-1250 C. This mineral assemblage is capable of taking large quantities of most radwaste elements in high-level wastes into solid solution. From electroprobe microanalysis of SYNROC melts Re 4+, Mo 4+, Rh 3+, and Fe 3+ are strongly partitioned into the hollandite phase. Rare earths and actinides prefer the zirconia phase, and perovskite, while strontium prefers perovskite. Cesium and rubidium enter "kalsilite" and leucite type phases. If recrystallized from the subsolidus these elements also will enter celsian and hollandite. SYNROC may be produced in a mechanically strong form with low porosity and low leaching rates. It is believed to be superior to borosilicate glasses under geological conditions. (NDV)

This paper was presented at the conference but was not published in the Proceedings.

WASTES, HIGH-LEVEL; CHEMICAL PROPERTIES; THERMAL PROPERTIES; SOLIDIFICATION; HOLLANDITE; ZIRCONIA; PEROVSKITE; CELSIAN; LEUCITE; LABORATORY STUDIES

<186>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Light-Water Reactor Wastes to be Isolated in Geologic Formations.

I/OWI/TR-88; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 3, (pp. 3-1 - 3-36), 837 pp.(1978, April)

Three alternative Light-Water Reactor (LWR) fuel cycles are considered as the basis for repository design: total recycle, spent

<106>

CONTAINER DESIGN AND PERFORMANCE

<106> CONT.
unreprocessed fuel (SUNF) cycle, and uranium-only recycle. A description of the waste contents and forms for each cycle is given as well as details in respect to the canisters, overpacks, waste drums, shipping containers and casks. The integrity of waste packing for a 5-year retrievability is discussed in addition to the projected waste receiving rates for each cycle. (CAJ)

Tabular data and references are included.

RECOMMENDATIONS; SITE SELECTION; WASTE STORAGE; WASTES, RADIOACTIVE; REPOSITORY; STORAGE, GEOLOGIC; WASTE MANAGEMENT; CONTAINMENT; SPENT FUELS; STRESS; CASKS; CONTAINER INTEGRITY; CONTAINERS; RETRIEVABILITY; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LOW-LEVEL; DRUMS

<107>
Shappert, L.B., D.S. Joy, and E.H. Heinkell, Oak Ridge National Laboratory, Oak Ridge, TN.

Radioactive Waste Transportation Systems Analysis and Program Plan.

ORNL-5362; 32 pp. (1978, March)

A systems analysis program has been developed by ORNL to identify and develop plans to deal with problems associated with the transportation of radioactive wastes to a federal repository in 1985. A logistics model has been developed to help determine the optimum shipping policies for low-level transuranic waste from generating sites to federal repositories. The model is currently capable of handling 15 waste generation sites, 10 federal repositories, and covers a 30 year time span. Several repository constraints, such as maximum receiving rates and total storage capability are also incorporated in the model. Optimality is evaluated by minimizing an objective function, such as system transportation costs, radiation exposure to the population, etc. Shipping containers for low-level transuranic waste from the defense program may not meet new NRC regulations (10 CFR 71.02) requiring two lines of containment following exposure to hypothetical accident conditions. New designs may be required before commercial waste can be shipped by rail to federal repositories. The current packaging has not been tested under accident conditions. (Auth) (LKH)

MODELS; TRANSPORTATION; CONTAINERS; WASTES, LOW-LEVEL; WASTES, TRANSURANIC; LEGISLATION; STANDARDS, FEDERAL; ACCIDENTS; HAZARD ANALYSIS; ECONOMICS

<108>
Smyrl, W.H., L.L. Stephenson, and J.W. Braithwaite, Sandia Laboratories, Albuquerque, NM.

Behavior of Candidate Canister Materials in Deep Ocean Environments.

SAND-76-9137; CONF-77[illegible]303; Corrosion 77/NACE Meeting, Proceedings of a Meeting, San Francisco, CA, March 14-18, 1977, (36 pp.). (1977)

Corrosion tests have been conducted under simulated deep ocean conditions for nine months. The experimental apparatus performed well, and indicates that longer term tests could be conducted. The materials chosen were titanium, zirconium and nickel base alloys, and all materials showed corrosion rates that were very low even at the highest test temperature. None of the materials showed susceptibility to stress corrosion cracking, and none gave evidence of differential aeration corrosion. Ambient tests confirmed the findings that none should be sensitive to differential oxygen effects. The zirconium alloys may be more susceptible to pitting corrosion than the others, although the pitting conditions are unlikely to be found in service, unless higher temperatures are encountered. It is concluded that all the materials tested are candidates for further study, and could give long life under deep ocean conditions. (Auth)

SEA DISPOSAL; MARINE SYSTEMS; CANISTERS; WASTES, RADIOACTIVE; CORROSION; LABORATORY STUDIES; TITANIUM; ZIRCONIUM; NICKEL; ALLOYS; ELECTROCHEMISTRY; STRESS; OXYGEN; PRESSURE EFFECTS

<109>
Smyth, J.R., R.J. Vidale, and R.E. Charles, Los Alamos Scientific Laboratory, Los Alamos, NM.

FIRM Wastes: Permanent Isolation in Rock-Forming Minerals.

Y/OWI/SUB-77/18268; CONF-770789; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 63-68), 92 pp. (1977, August 20)

The most practical system for permanent isolation of radioactive wastes in granitic and pelitic environments may be one which specifically tailors the waste form to the environment. This is true because if recrystallization of the waste form takes place within the half-lives of the hazardous radionuclides, it is likely to be the rate-controlling step for release of these nuclides to the ground-water system. The object of the proposed waste-form research at Los Alamos Scientific Laboratory (LASL) is to define a phase assemblage which will minimize chemical reaction with natural fluids in a granitic or pelitic environment. All natural granites contain trace amounts of all fission product elements (except Tc) and many contain minor amounts of these elements as major components of certain accessory phases. Observation of the geochemistry of fission-product elements has led to the identification of the natural minerals as target phases for research. Specifically we have proposed to experimentally determine the amounts of fission product elements which can stably be incorporated into selected mineral phases and to determine the leachability of the assemblage thus produced using fluids typical of the proposed environments at the Nevada Test Site. These phases include sphene, apatite, pollucite, and K-feldspar. This approach to waste isolation satisfies the following requirements: (1) It minimizes chemical reaction with the environment (i.e. recrystallization) which is likely to be the rate-controlling step for release of radionuclides to groundwater; (2) waste loading (hence temperature) can be easily varied by dilution with material mined from the disposal site; (3) no physical container

<149> CONT.
is required; (4) no maintenance is required (permanent); (5) the environment acts as a containment buffer. It is proposed that such wastes be termed PIRM wastes, for Permanent Isolation in Rock-forming Minerals.
(Auth) (LKN)

Only the abstract is given.

GRANITES; IGNEOUS ROCKS; STORAGE, GEOLOGIC; SEDIMENTARY ROCKS; SHALES; GEOCHEMISTRY; POLLUCITE; SPHENE; APATITE; FELDSPARS; CHEMICAL PROPERTIES; HYDROTHERMAL ALTERATIONS; WASTES, RADIOACTIVE; WASTE MANAGEMENT; WASTE DISPOSAL; LABORATORY STUDIES; WASTE-ROCK INTERACTIONS

<150>
Sobol, J., and T. Cooper. Westinghouse Advanced Power Systems Division, Madison, PA.

Radioactive Waste Management: The Need for Multiple Barriers.

CONF-780819; Nuclear Air Cleaning, Proceedings of the 15th DOE Conference, P.W. First (Ed.), Boston, MA, August 7-10, 1978, Vol. 2. (pp 998-1008), 1287 pp.(1979, February)

This paper examines isolation criteria for high-level radioactive waste disposal, in particular the "multiple barrier" concept. Based on Federal Maximum Permissable Concentration regulations for radionuclides in water, the essential period for waste containment is the first 1000 years; the multiple barrier system has the best chance of containing the wastes for such a period, and of meeting technical and public acceptance requirements for a waste management system. A realistic assessment of waste disposal effectivness must consider the leach-resistance of the waste form, engineered containment, and the absorption of the waste by the surrounding geologic media. Borosilicate glass waste forms have been produced in the U.S. with initial leach rates of 10(E-6) g/sq cm/day (leach time for 1 ft dia. X 10 ft long form would be 150,000 years); similar forms produced in France have initial leach rates of 10(E-7)-10(E-8) g/sq cm/day, and nepheline syenite glass waste forms from Canada have leached at 4 X 10(E-11) g/sq cm/day in ground water after remaining in place over 18 years. In Sweden, outer containers for the waste have been designed in the form of 100 mm of Pb overlain by 6 mm of Ti, the entire canister embedded in a bentonite-quartz sand mix to retard groundwater (flow rate reduction factors of 200 or more are calculated). Studies of geologic barriers have mainly emphasized salt, with its high thermal conductivity, radiation resistance, and plasticity, and the crystalline rocks, which are less corrosive and have a better public image. Studies by Battelle-Pacific Northwest Laboratory, which considered all the solidified high-level wastes produced by commercial power plant operations by the year 2000 placed in a repository in the Southwest, demonstrated that the maximum dose to an individual who obtained all his food and water from contaminated sources would be less than background. A similar study in Sweden for disposal in granite yielded a maximum individual dose of 13 mrem, assuming a defect in the engineered barrier. (LKN)

REPOSITORY; CONTAINMENT; CONTAINERS; LEACHING; STORAGE, GEOLOGIC; GLASS; VITRIFICATION; SYENITES; BENTONITES; GROUND WATER; SALT DEPOSITS; SALT DOMES; IGNEOUS ROCKS; RADIATION DOSE; GRANITES; WASTES, HIGH-LEVEL; WASTE DISPOSAL; WASTE STORAGE; WASTE MANAGEMENT; THEORETICAL STUDIES; REVIEWS

<151>
Sandquist, J.D., Corning Glass Works, Corning, NY.

Preliminary Corrosion Test of a Glass-Ceramic Candidate for a Nuclear Waste Canister.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp 289-296), 563 pp.(1979)

Preliminary corrosion studies of Corning Glass Works spodumene glass-ceramic Code 9617 as a storage canister for spent nuclear fuel rods indicates a corrosion rate of 0.013 cm/1000 yrs when stored in a granite repository at an initial temperature of 110 C, well within the targeted rate of 0.1 cm/1000 yrs. This study was part of the evaluation of a plan to encapsulate spent fuel packed with a mixture of sand and bentonite clay, and buried in deep shafts bored in granite. The main problem was to choose an encapsulation material that would be strong, sealable, and corrosion resistant over geologic time. Initial research on the chemical durability of glass-ceramic Code 9617 is presented in this paper. Several conditions of storage were specified: a) a 9:1 ratio of dry sand to clay packed around 800 spent fuel rods in a ceramic canister; b) the canister surrounded by the 9:1 mix in horizontal shafts 500 m deep in granite; c) the canister surface temperature at burial to be 100 C which would cool to 25 C after 3000 yrs; and d) ground water flow through the granite to be 1 l/sq m/yr. Accelerated tests using elevated temperatures or solutions of increased corrosivity are often used for practicality, but the method raises questions of validity of results. In this study accelerated tests were used, and a range was chosen for temperature, time, test solution, sample area, and the ratio (R) of test solution volume to sample surface area. Analytical techniques included weight loss, microscopy, hydrogen ion profiling, and flame spectrometric analysis for corrosion products. R was the most studied variable because it affects the rate of change of test solution chemistry with control and pressure as factors. Rather than accelerating corrosion and using repository conditions and extrapolate the data to a zero ratio (R) to give an estimate of the equilibrium corrosion rate under static (site) conditions. The corrosion rate for Code 9617 is estimated to be 0.013 cm/1000 yrs for static conditions. Limited tests indicated a negligible effect from the presence of sand and bentonite when added to the water solution. Since Code 9617 is a composite of several phases, it exhibits different localized corrosion rates; the glass phase is preferentially removed leaving the crystalline phase (90% spodumene, 6% rutile and spinel) untouched. Hydrogen or water migration into the surface of materials plays a major role in corrosion. Use of a recently developed method for determining hydrogen ion profiles allowed calculation of depth of etch, gel, and ion exchanged layers for water attack at different times and

<151> CONT.

temperatures. It was concluded that Code 9617 seems to have sufficient corrosion resistance to withstand ground water attack in a granite repository, and thus should be considered as a nuclear waste containment canister material. (Auth)(DCM)

Three tables provide the ranges of experimental variables, estimated glass-ceramic corrosion rates in water for 2 ratios, and depths of etch, gel, and ion-exchange layers. Two figures present the effect of R on the measured corrosion rate of Code 9617, and corrosion rate extrapolation to zero ratio.

Corrosion Rate

CONTAINMENT; ENCAPSULATION; GLASS; CERAMICS; DEPOSITORY; GRANITES; CANISTERS; STORAGE, GEOLOGIC; CORROSION; SPENT FUELS; LABORATORY STUDIES; TEMPERATURE; WASTES, RADIOACTIVE; WASTE DISPOSAL; WASTE STORAGE; CHEMISTRY; CHEMICAL PROPERTIES

<152>

U.S. Department of Energy, Division of Environmental Control Technology, Transportation Branch, Washington, DC.

Shipments of Nuclear Fuel and Waste...Are They Really Safe?.

DOE/EV-0008; 12 pp.(1977, October)

Presented is a DOE study on the safe transport of radwastes. Included in the discussion are the type of radwastes that are and will be shipped, the type of shipping cask required for the waste, and quality assurance in fabrication and use. Package integrity and packaging methods are also covered. From the study it appears that the probability of death, injury or massive property loss due to transportation of radioactive materials is determinable, and very small. The total accident probability for transportation of such material to and from nuclear power reactors and fuel storage sites, will be several orders of magnitude below the more common nonradiological causes. Various other studies indicate the likelihood of a catastrophic nuclear transport accident is so infinitesimal it can be said that one will never happen. Hence, the risk is probably acceptable. (NDV)

ACCIDENTS; CASKS; CONTAINERS; ECONOMICS; ENVIRONMENT; HAZARD ANALYSIS; PACKAGING; RADIATION HAZARDS; TRANSPORTATION; WASTE MANAGEMENT; WASTES, RADIOACTIVE

<153>

Weed, H.C., D.G. Coles, D.J. Bradley, R.H. Benning, and J.S. Schweiger, Lawrence Livermore Laboratory, Livermore, CA; Battelle-Pacific Northwest Laboratories, Richland, WA.

Leaching Characteristics of Actinides from Simulated Reactor Waste Glass.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 181-187), 563 pp.(1979)

Leaching characteristics of simulated reactor waste glass was determined by two experimental methods. Method one is single-pass leaching and method two is the one described by Mespo The first method was used at Lawrence Livermore Laboratory, and the study was based on statistically designed factorial experiments with unequal replication. Flow rate, temperature, leachant composition, and time are the factors or independent variables; and leaching rate is the response or dependent variable. The composition of the glass in mole % is SiO2-52, Na2O-16, B2O3-11, Fe2O3-5, MnO-5, TiO2-3, CaO-3, MoO3-1, and ZrO2-1. The radionuclides used in the experiment were Pu 239 and Np 237. Three leaching solutions were used, distilled water, a bicarbonate solution and a simulated brine. The flow rates varied from 300 cm cm/d to 10 cm cm/d, and the temperature ranged from 25 C to 75 C. The initial leach rate for Pu 239 during the first day is 8.4 X 10(E-7) with filters and 1.2 X 10(E-6) cm/d without filters. More studies are being done. (NDV)

Leaching Rate

Pu 239; Np 237

LEACHING; GLASS; PLUTONIUM; NEPTUNIUM; STATISTICS; METHODS; LABORATORY STUDIES; BRINES; WATER

<154>

Westra, A.G., J.S. Schofield, S.J. Horn, J.A. Hendricks, and G.S. Poniak, Idaho National Engineering Laboratory, Idaho Falls, ID.

Sampling of Stored High-Level Radioactive Calcined Waste at Idaho Chemical Processing Plant.

ICP-1186; 37 pp.(1979, May)

Equipment and procedures used to sample calcined solids stored in underground bins at the Idaho Chemical Processing Plant are described. The samples consisted of alumina calcine stored since 1967 and zirconia calcine stored since 1969. The purpose of the bin sampling program was to determine if the calcine had changed during the storage period. The main concern was that the calcine may have agglomerated; this would prevent pneumatic retrieval proposed for several long-term waste management alternatives. The calcine sampling process basically consisted of driving a string of sample tubes into a calcine storage bin with a soil sampling drill rig, and withdrawing the sample tubes one at a time into a shielded sample cask located directly over the open calcine retrieval riser. Samples of both alumina and zirconia calcine were obtained and indicate that the calcine can be retrieved with a pneumatic retrieval nozzle. The bin sampling was accomplished without contaminating the storage vault or the soil in the vicinity riser. Slight contamination was found in the anteroom of the Multiwire Cell after removing the testing equipment. The total penetrating radiation exposure for the whole operation was about 4.7 man-rem. (Auth)(NAF)

Several illustrations of the sampling process are given.

WASTES, HIGH-LEVEL; SAMPLING; CALCINATION; WASTE

<154> CONT.
STORAGE; ZIRCONIUM; ALUMINUM; CASKS;
RETRIEVABILITY; EQUIPMENT

<155>
Bentzik, J.M., Jr., and R.P. Tucotte,
Battelle-Pacific Northwest Laboratories,
Richland, WA.

Hydrothermal Glass Reactions in Salt Brine.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 341-348), 563 pp. (1979)

Leaching studies of nuclear waste glasses have usually been conducted at temperatures at or below 100 C. In this study temperatures of 250 C and 350 C were used to accelerate reaction rates and establish the reaction rates over a broad temperature range. A simulated high-level waste glass, a supercalcine, and some common ceramic and metallic solids were exposed to high-temperature salt brine and deionized water for time periods ranging from 3 days to 3 weeks. The glass underwent partial crystallization, depending on the conditions, and yielded NaFeSi2O6 as the primary alteration product. At the extreme temperature of these tests, cesium, rubidium, strontium, and molybdenum showed moderate to high solubilities in salt brine. Solubility in deionized water was an order of magnitude lower than in salt brine, but the glass alteration rate was higher. In a similar salt brine test, a supercalcine showed similar cesium and strontium releases when compared to the glass. (Auth) (DCW)

Two tables give a) leachability results at 350 C for glass and supercalcine in brine and deionized water, and b) results of scouting autoclave leach tests at 250 C for 8 different materials in 2 brines and deionized water.

Leaching Rate

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; CHEMICAL PROPERTIES; CHEMICAL COMPOSITION; LABORATORY STUDIES; BRINES; GLASS; TEMPERATURE; SOLUBILITY; CESIUM; RUBIDIUM; STRONTIUM; MOLYBDENUM; CERAMICS; LEACHING; EXPERIMENTS

<156>
White, E.L., R.E. Scheetz, D.M. Roy, K.G. Zimmerman, and H.W. Grutzeck, Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Permeability Measurements on Cementitious Materials for Nuclear Waste Isolation.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 471-478), 563 pp. (1979)

This study is concerned with permeability measurements on cementitious materials which might be used to plug boreholes and entrance shafts to nuclear waste geologic repositories. Accurate permeability measurement is of great importance in the search for materials that will seal repositories against inflow or escape of groundwater. The purpose of this paper is to show that routine permeability measurements on cementitious materials can be made over the range from millidarcys to microdarcys to tens of nanodarcys by utilizing nitrogen gas at a series of pressures and extrapolating to calculate an equivalent liquid permeability. The results are compared with those determined by measuring water permeability directly. Gas permeability measurements were made using a single medium-pressure cell or three cells manifolded together. This system utilizes a pressurized commercial nitrogen tank and regulator as the source of stable gas which can be varied from 18.7 to 800 psi (0.1 to 5.5 MPa). The gas which has passed through the sample is measured by the amount of water displaced in a calibrated cylinder. Water permeability was measured with a system that utilizes a commerical nitrogen gas cylinder and regulator as a source of stable pressure, generally maintained at 2000 psi (14 MPa) for these experiments. The gas pressure drives a piston which forces deionized water through the sample. The measured rate of water flowing through the sample, in ml/sec, is used to calculate the water permeability directly. Samples for which both water and gas permeabilities were measured represented a variety of compositions and curing conditions. For the most part, the reproducibility of the permeability value determined by those 2 different techniques was within a factor of 10, and more commonly within a factor of 3. Using the gas method, the liquid permeability of type 3 cement and quartz cured 2 months at room temperature was 1.35 X 10(E-5) darcys. Comparison with permeabilities obtained by the water method showed this value to be reasonable. Permeability measurements on cementitious materials using water as a medium posed potential problems related to hydration and leaching. The combination of the two techniques for determining permeabilities of cementitious materials has been shown to be useful. Each method has advantages and disadvantages, changing permeability, or damage caused to hydrated cement samples by the drying prior to gas permeability measurement. By combining measurements by the two methods, high confidence in the validity of the measurements was established. It is roughly estimated that in 2 X 10(E+5) years for a 27-inch (690 mm) diameter column, the amount of liquid transmitted through an initially saturated column at a unit hydraulic gradient would be about 2500 liters total, a very small amount. (Auth) (DCW)

Two graphs show gas permeability measurements extrapolated to infinite pressure for an equivalent liquid permeability value on a type 3 cement, and water permeability curves for five cement composites.

Permeability

WASTES, RADIOACTIVE; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; REPOSITORY; PERMEABILITY; GROUND WATER; CEMENTS; PHYSICAL PROPERTIES; LABORATORY STUDIES

<157>
Winchester, J.W., Florida State University,
Department of Oceanography, Tallahassee, FL.

<157>

CONTAINER DESIGN AND PERFORMANCE

<157> CONT.
long Term Geochemical Interactions of High Active Waste with Crystalline Rock Repository Media.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 385-387), 563 pp. (1979)

Safe long term isolation of high-level radioactive waste requires immobilization of most fission products on a thousand year time scale and of some heavy element alpha activity on a million year scale. For both groups a transport distance of the order of hundreds of meters has been considered to be the most relevant to nuclide migration from a deep geologic repository to points of human access. In conceptual repository designs, solid waste canisters would be emplaced a few meters apart in an array several hundreds of meters deep into bedded salt or crystalline rock, covering an area of a few square kilometers. Tens of thousands of canisters could be emplaced in a single repository. Emplacements would be connected by packed clay or silicate rock material filling. Geochemical redistribution within the repository itself, over distances measured in meters, could be important for the chemical stability of the repository, and chemical interactions between the emplaced canisters and the surrounding geologic environment could affect rates of radionuclide transport to man. Thus, the repository design should be optimized as a system simultaneously with respect to chemical processes both at short and long distances, as well as the thousand and million year time scales. If the ground water composition is altered by dissolving container materials, ion exchange retention of radionuclides by silicate minerals may be inhibited and the rate of radioactivity transport increased. Investigations on the effect of ion loading on retention of nuclides (Cs, Sr, Eu, Am) on montmorillonite show that radionuclide dispersion through geologic media surrounding a repository may be enhanced by high aqueous ion concentrations, especially of divalent ions from dissolving canisters. Conditions which inhibit large scale transport still may allow the waste to be redistributed locally, such as by ion exchange chromatographic effects creating zones of concentration within the repository. A repository system must be designed to preclude the concentration of fissile actinides into a critical mass. In a situation involving Am, Pu, and U, whose geochemical properties are distinct, creation of zones with high fissile content may occur. Studies have been done to calculate the fissile content in terms of minimum critical masses for 5 waste types, 4 geologic media, and decay times up to 100,000 years. Several examples illustrate the profound effects of different situations on the formation of a critical mass. (Auth) (DCW)

The studies and calculations referred to in this paper are from the 5 papers listed under references.

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, SOLID; STORAGE, GEOLOGIC; REPOSITORY; IMMOBILIZATION; CONTAINMENT; CANISTERS; FISSION PRODUCTS; ACTINIDES; RADIONUCLIDE MIGRATION; RADIONUCLIDES; SALT DEPOSITS; SILICATES; HOST ROCKS; GEOLOGY; WASTE-ROCK INTERACTIONS; GROUND WATER; STABILITY, CHEMICAL; CHEMISTRY; GEOCHEMISTRY; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; ADSORPTION; MINERALS; ION EXCHANGE; CRITICAL MASS; STRONTIUM; CESIUM; EUROPIUM; AMERICIUM; PLUTONIUM; URANIUM; CURIUM

<158>
Wodrich, D.D., Atlantic Richfield Hanford Company, Richland, WA.

Retrievable Surface Storage of High-Level Radioactive Wastes.

Transactions of the American Nuclear Society 17:326; CONF-731101; American Nuclear Society Winter Meeting, San Francisco, CA, November 11-15, 1973, (1 p.). (1973)

Since ultimate disposal methods for high-level radioactive wastes are not expected to be available before 1983, the AEC plans to build an interim storage facility to serve until a permanent repository has been established. This Retrievable Surface Storage Facility (RSSF) will provide storage for all high-level waste generated through 2000 AD. Wastes will be packaged in steel canisters 1 ft in diameter by 10 ft long, each containing the residue from 3 MT of reprocessed fuel and generating about 5 kW of decay heat. It is estimated there will be 75,000 such canisters by 2000 AD. Alternative storage concepts for the RSSF include cooled-water basins, air-cooled vaults, or air-cooled steel cylinders stored outdoors. Preliminary studies indicate that a steel cask with 14 in. walls (weight 30 tons) would reduce gamma radiation from a typical waste canister to below 5 mR/hr at the surface. This concept appears feasible, safe, and economically competitive with the other concepts under consideration. An attractive feature of this concept is that expenditure of capital funds for cask procurement can be delayed until needed. (LKH)

STORAGE, ABOVEGROUND; WASTES, HIGH-LEVEL; CANISTERS; CONTAINERS; DESIGN; COOLANTS; THEORETICAL STUDIES; WASTE STORAGE; WASTE MANAGEMENT; ECONOMICS

<159>
Wranglen, G., Royal Institute of Technology, Stockholm, Sweden.

Gold and Selective Storage Solves Nuclear Waste Problem.

Annals of Nuclear Energy 4:527-528. (1977)

In a nuclear reactor, nuclear fuel (uranium) is transformed into nuclear ash (fission products) and transuranics (including plutonium). After decay of the nuclear ash, the spent fuel still has a very high toxicity [10(E+3)-10(E+4) times as high as that of the nuclear ash per Ci on inhalation], of which the alpha-activity may be taken as an approximate measure. Because of this, direct disposal of spent fuel is not a practical or safe solution of the nuclear waste problem and for isolation of the transuranics, a permanent water-tight encapsulation is required. It is suggested that the use of a thick layer of gold plate on the disposal canisters be used to accomplish this. Because of obvious economic problems, the spent fuel must be reprocessed to extract the uranium and plutonium, leaving only a small

<159> CONT.

amount of plutonium and the remaining transuranics (50% Np and 20% Am). These wastes would amount only to 700 gm of metal or 800 gm of oxide with a volume of 100 cubic centimeters, including an adequate amount of neutron poison (e.g. gadolinium) to exclude criticality per ton of uranium. For such a small volume, the transuranic waste can be enclosed in relatively small, short cylinders of stainless steel, alloyed with boron and lined with cadmium and lead sheets to eliminate radiolysis. These cylinders are then coated with a 1 mm (40 mil) layer of pure gold by electrodeposition. The required amount of gold is estimated to be 60 gm/t of uranium used as fuel at a cost of less than 0.001 cents per kwh. On top of the gold coating a thick layer of copper is applied as a mechanical protection for the gold coating. As a final coating the waste canisters are enclosed in a thick layer of bitumen and permanently disposed at great depth in a suitable geologic formation. In the unlikely case that the bitumen is destroyed, the copper layer will slowly corrode away, but the corrosive attack will stop at the gold layer. (JWT)

BITUMENS; CONTAINERS; COPPER; GOLD; LEAD; CADMIUM; STEELS; ELECTRODEPOSITION; ASHES; ALLOYS; BORON; CORROSION; FISSION PRODUCTS; FUEL ELEMENTS; FUEL REPROCESSING; COST BENEFIT ANALYSIS; AMERICIUM COMPOUNDS; NEPTUNIUM COMPOUNDS; PLUTONIUM; URANIUM; TRANSURANICS; VOLUME REDUCTION; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE; WASTES, TRANSURANIC; CONTAINER INTEGRITY

<160>

Tarzwell, R., S. Konieczny, and B. Barraclough, Los Alamos Scientific Laboratory, Los Alamos, NM.

Experimental Studies of the Degradation of RAD-Wastes for the Sandia Laboratories Waste Isolation Pilot Project (WIPP), April 1-June 30, 1978.

LA-7478-PR; 5 pp.(1978, October)

Forty-three experiments are in progress in which simulated waste matrices contaminated with plutonium 239 oxide (PuO_2) are contained in 300 cc cm stainless steel cylinders at 1 atm pressure. There has been no measurable radiolytic gas evolution from any of materials. However, some thermal degradation of paper at 78 and 100 degrees C has occurred. An experiment involving radiolytic degradation of cellulose at WIPP lithostatic pressure (150 atm) is in progress. (JWT)

LABORATORY STUDIES; PLUTONIUM; CONTAINERS; CASKS; WIPP; WASTES, TRANSURANIC; DEGRADATION; THERMAL PROPERTIES

<161>
Albrecht, E., and F. Perzl, Gesellschaft fur Strahlen-und Umweltforschung Munchen, German Federal Republic.

The Research and Development Program on Waste Disposal in the Federal Republic of Germany.

Waste Management and Fuel Cycles '78, W.G. Post and M.E. Wacks (Eds.), Proceedings of a Symposium, Tucson, AZ, March 6-8, 1978. Arizona Board of Regents, (pp. 239-251), 661 pp. (1978)

The research and development on waste disposal in Germany is reviewed with particular attention on the salt deposits and the Konrad Iron Mine. Safety aspects in using the salt deposits, in particular the Asse Salt Mine, is the solidification of the waste, the impermeable halite formation, and the several hundred meters of overburden strata. For the past ten years the aim at the Asse Pilot Project is the development, testing and optimization of transport systems and disposal technologies for low-level wastes. For the past five years similar work has been done for intermediate-level waste. Experiments have been carried out with electric heaters to simulate heat generation by high-level glasses. The results compare well with theoretical heat calculations. Construction of other caverns has begun for intermediate-level wastes. A feasibility study, geologic and mechanical, of the rocks of the Konrad Iron Ore Mine was done. Future tanks were also outlined. (NDW)

DIAPIRS; HALITE; CAVES; CONTAINER INTEGRITY; DISPOSAL SITE; FIELD STUDIES; GEOLOGY; REPOSITORY; SALT DEPOSITS; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL

<162>
Atlantic Richfield Hanford Company, Research and Engineering Division, Research Department, Richland, WA.

241-T-106 Tank Leak Investigation.

ARH-2978; 47 pp. (1973, November)

On June 8, 1973, the 241-T-106 underground liquid waste storage tank, located on the U.S. Atomic Energy Commission's Hanford Reservation, was confirmed as leaking. Approximately 835 cu m of radioactive liquid waste leaked into the sediments adjacent to the tank. It was predicted that the leaked waste would be retained by the dry sediments well above the water table. To confirm this prediction, 16 wells were drilled to delineate the contaminated zone. The deepest penetration observed in the wells drilled was 27 m below ground surface or 15 m above the water table. An assessment of the relative movement of radionuclides from the T-106 tank based on gamma activity analyses showed plutonium with the least mobility, followed by cesium, strontium, europium, cerium, antimony, cobalt, and ruthenium. The 1 uCi/liter ruthenium isopleth was used to delineate the contaminated zone. The volume of sediments involved was calculated to be about 25,000 cu m. Four wells, drilled north and east of the 241-T-106 tank, encountered leakage of liquid waste resulting from a failed grout seal in a sprue fill line. Using the 1 uCi/liter Ru 106 isopeth, the volume of sediments involved in this leak was estimated at about 1700 cu m. The greatest depth to which this liquid waste penetrated was about 25 m below the ground surface, or about 37 m above the water table. The results of this study confirm the predictions made soon after the leak was discovered. Based on these results and the basic knowledge of liquid movements in Hanford sediments established since the start of plant operations, further movement of radioactivity from its present location will be negligible. (Auth) (LKM)

Ru 106

ACCIDENTS; TANKS; LEAKS; WASTES, LIQUID; WASTES, RADIOACTIVE; RADIONUCLIDE MIGRATION; MONITORING; INFILTRATION; SOILS; SEDIMENTS; WATER TABLE; VADOSE ZONE; WELLS; FIELD STUDIES; DISPOSAL SITE; WASTE DISPOSAL; WASTE STORAGE

<163>
Borg, I.Y., R. Stone, H.B. Levy, and L.D. Ramspott, Lawrence Livermore Laboratory, Livermore, CA.

Information Pertinent to the Migration of Radionuclides in Ground Water at the Nevada Test Site - Part 2: Annotated Bibliography.

UCRL-52078 Part 2; 112 pp. (1976, August 31)

This report consists of the bibliography and abstracts that were compiled in the course of searching the literature for information on the migration of radionuclides at the Nevada Test Site. The abstracts are designed to be useful primarily to current and future investigations at NTS. Data that can be extrapolated to the NTS situation are reported, to the exclusion of those that cannot. A "relevance factor" of 1, 2, or 3 has been assigned to each item abstracted, the most relevant being 1, and the least relevant, 3. The bibliography consists of three sections: A. General Hydrology and Hydrogeology of the Nevada Test Site and Other U.S. Sites of Nuclear Explosions (58 references); B. Distribution and Transport of Radioactivity (109 references); and C. Sorption and Leaching Studies (95 references). Although the general subject of the effects of underground nuclear explosions is relevant to issues associated with radionuclide migration, the effects literature was not abstracted for the present work. This bibliography cannot be considered exhaustive. Searching was done with the ERDA/RECON on-line computer information retrieval system developed by Oak Ridge National Laboratory using as data sources Nuclear Science Abstracts, Water Resources Abstracts, Nuclear Safety Information Center, Engineering Index, and Energy Data Base (EDB). On-line searching was also carried out using GEOREF of the American Geological Institute. In addition, manual searching was done in Nuclear Science Abstracts for papers prior to 1962. Several existing bibliographies proved useful for scanning the foreign literature. (Auth) (LKM)

BIBLIOGRAPHIES; RADIONUCLIDE MIGRATION; DISPOSAL SITE; HYDROLOGY; GEOLOGY; EXPLOSIONS, NUCLEAR; SORPTION; LEACHING; COMPUTERS

<164>
Chapin, J.A., and F.E. Hind, EG and G Idaho,

<164>

DISPOSAL SITE

<164> CONT.
Inc., Idaho Falls, ID.

Burial Grounds.

TREE-1250 (Volume 2); Decontamination and Decommissioning, Long Range Plan, Idaho National Engineering Laboratory, C.R. Tolman (Ed.), Appendix D, (pp. D-1 - D-39), 319 pp.(1978, June)

The Radioactive Waste Management Complex, situated in a shallow valley comprised of shallow sediments and the Snake River Basalt, is composed of the Subsurface Disposal Area (SDA), Transuranic Disposal Area (TDA), Decontamination Facility South (DFS), Transuranic Storage Area (TSA), Intermediate-Level Transuranic Storage Facility (ILTSF), and Waste Volume Reduction Facility (WVRF). Subsurface and aboveground disposal in pits and trenches is carried on in the SDA. The pits hold low-level beta-gamma wastes, while the trenches are used for disposal of high-level beta-gamma wastes. Aboveground storage of TRU wastes is accomplished at TDA and retrievable storage is done at TSA. Approximately 26,000 cu m of TRU-contaminated waste has been retrievably stored. Special burial grounds are also mentioned. Contaminated materials resulting from an accidental excursion at the SL-1 nuclear reactor were buried in two pits and one trench in the SL-1 burial ground. A similar burial ground exists for the reactor remnants of the BORAX-I reactor which was intentionally subjected to a rapid power excursion in July 1954. Most of the fuel plates were melted and there was gross damage to the reactor. High-level liquid radwastes are stored at the ICPP tank farm before being solidified at the Waste Calcining Facility and placed into retrievable storage. The tank farm is comprised of 11 tanks with a capacity of 1136 cu m each and 4 tanks with capacities of 114 cu m. For high-level solids three underground concrete vaults have been pressed into service. The first two are filled and the third is presently being filled. The first facility has 4 bins with a total capacity of 221 cu m; the second has a total capacity of 114 cu m. The third facility is essentially the same as the second. Fuel from the Peach Bottom HTGR Core 1 reactor will be stored for 10 years before processing. The Graphite Fuel Storage Facility is used to store spent fuel from Fort St. Vrain and Peach Bottom (Core 2) HTGR reactors and temporary storage for fuel from the Rover nuclear rocket program. Other storage facilities and the history of the Complex are included. (DEV)

ACCIDENTS; BETA PARTICLES; BURIAL; DISPOSAL SITE; RADIATION, GAMMA; MONITORING; PITS; REACTORS; REPOSITORY; STORAGE, ABOVEGROUND; STORAGE, GEOLOGIC; SOLIDIFICATION; TANKS; WASTE MANAGEMENT; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, TRANSURANIC; WASTES, SOLID

<165>
Cohen, P., and C. Gaillodreau.

Preliminary Investigations of the Adsorption of Radiostrontium in Saclay Soil. Three-Component System.

Ground Disposal of Radioactive Wastes, H.J. Roofman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 98-99), 168 pp.(1961, July)

A summary of measurements of Sr 90 partition coefficients in Saclay soil, and a preliminary study on adsorption of trace ions from a solution containing two bulk ions are presented in this document. The Saclay soil contains a clay mixture of 45% montmorillonite, 45% kaolinite and 10% illite, and exhibits a cation exchange capacity of 0.21 meq/g. Studies of two and three component systems are discussed. The bicomponent systems considered are Sr-K, Sr-NH4, and Sr-Na. The three-component system discussed is the Na-K-Cs system. The partial partition coefficients equations are fully presented for both systems. (PTO) (MDV)

Ion Exchange Capacity

SOILS; CLAYS; ABSORPTION; IONS; MONTMORILLONITE; KAOLINITE; ILLITE; ION EXCHANGE CAPACITY; CATIONS; SODIUM; POTASSIUM; STRONTIUM; CESIUM; EQUATIONS; MASS BALANCE

<166>
Colombo, P., A.J. Weiss, and A.J. Francis, Brookhaven National Laboratory, Department of Nuclear Energy, Nuclear Waste Management Research Group, Upton, NY.

Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites.

BNL-NUREG-50855; Progress Report No. 6 July-September 1977; 44 pp.(1978, May)

A survey of the Maxey Flats, Kentucky, low-level radioactive waste disposal site was conducted to obtain an overview of the radioactivity in the trench waters of the purpose of selecting specific trenches for comprehensive study. Water samples collected from trenches and wells were analyzed for specific conductance, pH, temperature, dissolved organic carbon, tritium, gross alpha, gross beta, and gross gamma radioactivities. The results indicate that there are large differences in the composition of trench wastes at the site. Values of specific conductance ranged from 400 to 38,500 micro ohs/cm at 25 C. The p of the samples varied from p 2.2 to 12.4 indicating differences in chemical composities of the trench waters. Gross alpha activity ranged from <6(E+0)-6.39(E+5) pCi/l; gross beta activity from <3(E+1)-1.77(E+6) pCi/l; gross gamma activity from <1(E+1)-6.20(E+3) pCi/l; and tritium from 1.80(E+3)-8.67(E+9) pCi/l. Dissolved organic carbon ranged from 7.4 mg/l to 5820 mg/l. Several trenches that represent extreme and average values of the major parameters measured have been tentatively selected for further study. (Auth) (MAF)

Specific Conductance; pH; Temperature

N 3

WASTES, LOW-LEVEL; TRENCHES; WATER; SAMPLES; WASTE DISPOSAL; RADIONUCLIDES; ORGANIC COMPOUNDS; CARBON; TRITIUM; ALPHA PARTICLES; BETA PARTICLES; RADIATION, GAMMA; CHEMICAL ANALYSIS; WELLS; SCINTILLATION COUNTERS; STREAMS; SITE SURVEILLANCE

DISPOSAL SITE

<167>
Colombo, P., A.J. Weiss, and A.J. Francis, Brookhaven National Laboratory, Department of Nuclear Energy, Nuclear Waste Management Research Group, Upton, NY.

Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Low-Level Radioactive Waste Disposal Sites.

BNL-NUREG-50739; Progress Report No. 5 April-June 1977; 39 pp. (1977, December)

Water samples were collected from ten trenches at the Maxey Flats, Kentucky, low-level radioactive waste disposal site and analyzed for the presence of organic compounds. Identification of the organics was achieved by combined gas chromatography/mass spectrometry. The concentration of dissolved organic carbon varied from trench to trench and was generally in the range of 200 to 8,500 mg/l. Organic compounds found in the samples were predominantly organic solvents, decontamination liquids such as tributylphosphate, alcohols, aliphatic and aromatic acids, and several unidentified adipates and phthalates. In addition, volatile amines such as cyclohexylamine and dicyclohexylamine were also identified. The organic compounds may play an important role in the formation of organo-metallic complexes, leachability, solubility, and mobility of the radionuclides that are buried in the trenches. (NAP)

ORGANIC COMPOUNDS; TRENCHES; WATER; SAMPLES; SITE SURVEILLANCE; SPECTROMETRY; WASTES, ORGANIC; CARBON; BURIAL; SEPARATION PROCESSES; CHEMICAL ANALYSIS; ACIDS, ORGANIC; MICROORGANISMS

<168>
Colombo, P., A.J. Weiss, and A.J. Francis, Brookhaven National Laboratory, Department of Nuclear Energy, Process Technology Division, Upton, NY.

Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites.

BNL-NUREG-50695; Quarterly Progress Report January-March 1977; 20 pp. (1977, September)

Water samples were collected from ten trenches at the Maxey Flats, Kentucky, disposal site in September 1976. Precautions were taken to collect and process the samples under anoxic conditions in accordance with procedures developed for this purpose at Brookhaven National Laboratory. The methods used for the preparation of samples and the analytical techniques employed for inorganic and radiochemical analyses are described. Inorganic analyses included field measurements of pH, temperature, specific conductance, and color. Dissolved metals measured were Ca, Fe, Li, Mg, Mn, K, and Na. Non-metals measured include chloride, nitrate, nitrite, silica, and sulfate. Analytical results of the dissolved radiochemical constituents indicate that all trench water samples contain tritium, Sr 90, and Pu isotopes. The concentrations for Pu 238 ranged from 5.0-1.7 X 10(E+5) picocuries/liter and for Pu 239, 240 from <1.0-2.1 X 10(E+0) picocuries/liter. Cobalt 60 and Cs 137 were also detected in most trench water samples. The concentrations of the dissolved inorganic and radiochemical constituents are tabulated for each trench. (Auth) (NAP)

pH; Temperature; Specific Conductance

Sr 90; Co 60; Cs 137; Am 241; Pu 239, 240; Pu 238

WATER; SAMPLES; SITE SURVEILLANCE; CHEMICAL ANALYSIS; RADIOCHEMISTRY; TRITIUM; STRONTIUM 90; PLUTONIUM 238; PLUTONIUM 239; AMERICIUM 241; COBALT 60; CESIUM 137; INORGANIC COMPOUNDS; CALCIUM; IRON; LITHIUM; MAGNESIUM; MANGANESE; POTASSIUM; SODIUM; CHLORIDES; NITRATES; NITRITES; SILICA; SULFATES; ALKALINITY

<169>
Comptroller General of the United States, Washington, DC.

Issues Related to the Closing of the Nuclear Fuel Services, Incorporated, Reprocessing Plant at West Valley, New York.

HRD-77-27; Report to the Conservation, Energy, and Natural Resources Subcommittee, House Committee on Government Operations, 34 pp. (1977, March 8)

When the Nuclear Fuel Services Reprocessing Plant at West Valley, NY was shut down for enlargement and reduction to radiation exposure by the employees, a number of questions were raised about the operation of such a plant. The site consists of 4 high-level storage tanks containing 612,000 gallons of waste, a high-level burial ground containing about 100,000 cu ft and a low-level burial ground containing about 2 million cu ft of solid radwastes. The tank integrity is in doubt, and before the wastes can be extracted and solidified the tank condition must be verified. There is also a question as to whether or not the tanks could withstand an earthquake. Since there has been a floatation problem below the tanks and vault system the soil characteristics should be determined in case of a tank rupture. A priority item is the characterization of the physical and chemical properties of the high-level waste sludge for satisfactory extraction. No solidification process has been developed for managing the wastes. The government will have to provide technical assistance to NFS and New York for studying and correcting many of the site problems. Also the government will have to establish guidelines and plans for 1) handling the spent fuel stored and buried, and decommissioning the reprocessing plant and 2) long term care requirements for the high- and low-level waste burial sites. A policy of assistance to New York State should be developed with the Federal government. (NDV)

BURIAL; CONTAINER INTEGRITY; CONTAMINATION; COST BENEFIT ANALYSIS; DECOMMISSIONING; DISPOSAL SITE; FUEL ELEMENTS; HAZARDS; LEGISLATION; PLANTS, WASTE TREATMENT; SITE EVALUATION; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE

<170>
Crawley, D.T., Atlantic Richfield Hanford Company, Richland, WA.

Plutonium-Americium Soil Penetration at 234-5 Building Crib Sites.

<170>

DISPOSAL SITE

<170> CONT.
BNW-1278; 6 pp.(1969, June)

Core samples were collected from 6 shallow dry wells at the 216-Z-1A and 216-Z-12 waste disposal sites to determine the depth of penetration of Pu and Am in the soil of the 230-5 Building crib sites. Samples were taken at approximately 2-foot intervals and analyzed for Pu using a Ge(Li) detector. The Z-1A tile field has been used for disposal of high-salt solution wastes from the Pu reclamation facility via specific retention in the soil column. Pu concentration in samples from this area ranged from 0.6 X 10(E-8) g Pu/g soil at 18 feet to 12.8 X 10(E-8) g Pu/g soil at 10 feet. Based on sample number 2 from Z-1A, calculations show a soil liquid Pu concentration of 6.3 X 10(E-3) g/L. This is 1.2 times that of the effluent stream, indicating that there is a concentrating mechanism in the soil. Pu was present to the depth sampled (19 feet); coring was discontinued at this depth to avoid release of contaminants to ground water. Use of this field has been discontinued to allow the liquid to come to equilibrium with the soil. Samples from Z-1A show no rapid decrease in Pu content with depth, while those from Z-12 do. At Z-12, a concentration of 5 X 10(E-8) g Pu/g soil was detected at a depth of 2 feet below the crib, but at 8 feet and below, no Pu was detectable. Approximately 25 column volumes of liquid have passed through the used portion of the Z-12 crib. (LKM)

Pu

CORES; WELLS; PLUTONIUM; ADSORPTION; CRIBS; SALTS; CONCENTRATION; RADIOACTIVITY; SOILS; FIELD STUDIES

<171>
Erdal, B.R. (Ed.), et al., Los Alamos Scientific Laboratory, Los Alamos, NM.

Waste Management Programs and Capabilities within the Los Alamos Scientific Laboratory.

LA-7580-MS; 57 pp.(1978, October)

This report summarizes current radioactive waste management programs and related areas at LASL, for the purpose of aiding in the evaluation and coordination of these programs, to locate areas where additional work can be done, to provide a basis for development of new proposals, and to help other agencies assess the current and potential contributions of the laboratory to waste management research. Programs cover the areas of disposal site selection, the geologic-geochemical environment, mathematical modeling, field testing and model verification, detection and monitoring, systems analysis, and waste handling, processing and recycling. Divisional facilities exist for the carrying on of work in applied photochemistry, chemistry-materials science, nuclear chemistry, the geosciences, health research, field testing, energy, systems analysis and assessment, and theoretical studies. (LKM)

GEOLOGY; GEOCHEMISTRY; MODELS, MATHEMATICAL; THEORETICAL STUDIES; WASTE MANAGEMENT; SITE SELECTION; WASTE TREATMENT; MONITORING; WASTES, RADIOACTIVE

<172>
Fernandez, S.J., G.B. Pierce, and R.G. Botes, Exxon Nuclear Idaho Company, Inc., Idaho Falls, ID.

A Continuous Tritium Monitor for Nuclear Facility Off-Gas Streams.

ENICO-1022; 27 pp.(1980, January)

The continuous monitoring of tritium in stack effluents from nuclear facilities such as the Idaho Chemical Processing Plant (ICPP), is a difficult analytical problem due to: (1) tritium's low beta energy (beta max. = 0.018 MeV), and (2) the presence of relatively large concentrations of Kr 85 and C 14. Existing techniques were reviewed and a new monitor proposed for development. This monitor catalytically oxidizes all forms of tritium to HTO; separates the HTO from Kr 85 and C 14 by permeation distillation, and measures on-line the H 3 in a purified sample stream. In the laboratory, a suitable catalytic oxidizer was identified and the associated kinetic expressions, which allow its adaption to specific sampling conditions, were developed. The equations for calculating the efficiency and selectivity of a permeation distillation dryer, used to separate the HTO from the Kr 85 and C 14, were also determined. With a commercial permeation distillation dryer (permatube), quantitative tritium recovery was demonstrated and H 3 separation factors from Kr 85 and C 14 were measured. The respective Kr 85 and C 14 separation factors were 2.2 X 10(E+4) and 200. A 20-l ionization chamber was selected for use as the on-line H 3 detector, since in view of the large separation factors, additional selectivity was not required. A detection limit of 10(E-9) uCi H 3/cc cm, approximately background H 3 level, was established for the monitor and no significant memory effects were observed. The in-plant demonstration established that no bias exists between the presently used standard silica gel analysis technique and the continuous H 3 monitor. In addition, it was demonstrated that plant process transients were adequately reflected in the response of the monitor. The monitor is deemed suitable for use at nuclear fuel reprocessing plants. (Auth)

H 3; Kr 85; C 14

MONITORING; TRITIUM; EFFLUENTS, AIRBORNE; DISPOSAL, STACK, DESIGN; PLANTS, WASTE TREATMENT; EQUIPMENT; METHODS; INSTRUMENTS; MEASUREMENTS

<173>
Fickies, R.N., and R.H. Fakundiny, New York State Education Department, New York State Geological Survey/State Museum, Albany, NY.

Geotechnical Analysis of Soil Samples from Test Trench at Western New York Nuclear Service Center, West Valley, New York.

NUREG/CR-0640; NYSGS/79-2001; 21 pp.(1979, April)

In July 1977, a deep research trench was excavated and soil samples collected at the Western New York Nuclear Services Center, West Valley, New York. The glacial till horizons sampled are considered to be

<173> CONT.
representative of the till serving as a burial medium at the nearby low-level radioactive waste burial ground. A series of laboratory tests were conducted consisting of unit weight, moisture content, Atterberg limits, unconfined compression, dispersion, swell, permeability, and consolidation. These laboratory analyses and field observations indicate that the till exposed in the research trench is a generally dense mixture of silt and clay of low to medium plasticity, with minor amounts of fine to coarse sand and fine gravel. The till has a generally low coefficient of permeability in the range of 10(E-7)cm/sec horizontal and 10(E-8)cm/sec vertical. A network of vertical fractures exists in the upper 15 feet of "weathered" till which may allow some downward percolation of surface runoff. The test data indicates that the maximum depth to which these fractures could possibly penetrate is 50 feet. (Auth)

Moisture Content; Plasticity; Permeability Coefficient

SOILS; SAMPLES; TRENCHES; EXCAVATION; MORPHOLOGY; TILL; BURIAL; WASTES, LOW-LEVEL; LABORATORY STUDIES; FIELD STUDIES; MOISTURE; PRESSURE EFFECTS; DISPERSION; PERMEABILITY; SILTS; CLAYS; PLASTICITY; SANDS; GRAVELS; SEEPAGE FRACTURES; WEATHERING; PERCOLATION; RUNOFF

<174>
Gsirk, F.F., E.B. Hoskins, W.G. Pariseau, J.E. Bessell, and W.B. Haversik, RE/SPEC, Inc., Rapid City, SD.

Analysis and Evaluation of the Rock Mechanics Aspects of the Proposed Salt Mine Repository.

CONL/SUB-3706/1; Summary Progress Report RSI-0002; 21 pp.(1972, June 30)

A comprehensive analysis and evaluation of the rock mechanics aspects of a proposed salt mine repository are presented. Determined in the study were the thermal elastic-plastic and thermal viscoelastic states of stress and deformation resulting from simulated excavation and fill procedures, and fracture potential of the rock strata during simulated use of the repository. The salt the tests and the subsequent computer analyses were done on cores from the Hutchinson Shale and the Hutchinson Salt Formations in Lyons, Kansas. Tests were conducted to determine the tensile and compressive strengths, the modulus of elasticity, Poisson's ratio, and the steady-state creep parameter under room temperature and atmospheric pressure. Results indicate that the moduli of elasticity of the Stone Corral and members below the Hutchinson Salt are about an order of magnitude greater than those of the Harper Sandstone, Hinnescah Shale, and Hutchinson Salt. This suggests an alternating sequence of elastically "hard" and "soft" layers. Compressive strengths for the Harper Sandstone, Hinnescah Shale and the Hutchinson Salt are about 2,500 psi with tensile strengths between 220 and 800 psi. Compressive and tensile strengths of the Stone Corral are about 3,500 and 500 psi, respectively. The creep tests suggest that transient creep in both shale and salt can be described by a power function of time and the time-dependent deformations in shale and salt are due to different mechanisms. Preliminary elastic-plastic analyses indicate that stress concentration values of two, coupled with compressive strength values of 2,000 psi (salt), will give rise to incipient failure and plastic flow at a depth of 1,000 ft. Preliminary results of the viscoelastic analyses show the relaxation of the stresses in a pillar gives rise to a horizontal tension stress in the rib which may be significant. It is recommended that the material property data for a site be determined during the initial stages of site selection and development so the best mining system may be designed. (NDV)

DESIGN; EXCAVATION; FRACTURES; DEFORMATION; ROCK MECHANICS; COMPUTER PROGRAMS; LABORATORY STUDIES; MINING; MODELS; REPOSITORY; SALT DEPOSITS; SANDSTONES; SHALES; STRENGTH, COMPRESSIVE; STRENGTH, TENSILE; CREEP; STRESS; VISCOELASTICITY; ELASTOPLASTICITY

<175>
Harmon, K.M., Battelle-Pacific Northwest Laboratories, Richland, WA.

Disposition (D and D) of Retired Contaminated Facilities at Hanford.

BNWL-2126; Nuclear Waste Management and Transportation Quarterly Progress Report, April Through June 1976, L.H. Platt, Section 3.0, (pp. 3.1-3.6), 48 pp.(1976, October)

This program has the following objectives for FY 1975-76: to establish methods, costs, and priorities for the disposition of retired (excessed) Hanford facilities; and to identify and perform required facility disposition R&D. Analyses of surface crust and deposits in the reactor retention basin pipelines at Hanford indicate alpha contamination ranges of 20 to 400 pCi/g. The lines will have to be cleaned before they can be released. The deposits in all effluent lines except that from the KE Reactor were very loose, suggesting that a high-pressure water jet may be the best means of cleaning. The KE Reactor effluent line may require mechanical or chemical methods of decontamination. The gamma-emitters found in the deposits are mainly Eu 152, Eu 154, Eu 155, and Co 60. Cesium 137 was found occasionally, and Zn 65 and Sb 125 were found in the KE Reactor effluent line deposits. Gamma radionuclide concentrations were 5 X 10(E-4)uCi/g to 2 uCi/g. Decontamination and decommissioning scenarios for Hanford have been redefined. The following four cases are under current valuation: Case 1. prepare all retired sites so that unrestricted use can be permitted within 100 years; Case 2. prepare all retired sites so that unrestricted use can be permitted within 500 years - complete all D and D project activities within 100 years; Case 3. prepare selected sites for long-term safe storage of their radionuclide inventory; prepare all other sites so that unrestricted use can be permitted within 500 years; complete all D and D projects within 100 years; and Case 4. provide indefinite-term maintenance, surveillance, and access control to all sites. Certain decommissioning modes which fit these scenarios are suggested. Preliminary cost estimates and relative D and D priorities for selected facilities are presented. A preliminary priority listing, which indicates the relative urgency for undertaking some type of D and D action with each facility, has been prepared for 366 inactive or standby contaminated ERDA

<175>

DISPOSAL SITE

<175> CONT.
facilities at Hanford. Radiological criteria governing the release and use of decommissioned nuclear facilities were updated for the 107-DR Reactor Retention Basin System in accord with recent changes in release fraction methodology, and criteria were calculated for the 216-S-1, 2 crib. A prototypic water cannon, for use in spalling contaminated concrete surfaces, has been assembled by Exotech Inc., tested, and shipped to PNL for further evaluation. (LKH)

Eu 152; Eu 154; Eu 155; Co 60; Cs 137; Zn 65; Sb 125

DECOMMISSIONING; METHODS; RETENTION PONDS; REACTORS; DECONTAMINATION; COST ESTIMATES; EQUIPMENT; EVALUATION; RADIONUCLIDES

<176>
Barlsson, D.H. (Editor), EG and G Idaho, Incorporated, Idaho Falls, ID.

Safety Review Document for the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory.

IDFF-1060; 153 pp.(1977, May)

This report discusses the historical background of radioactive waste management at INEL, physical characteristics of the site, the Radioactive Waste Management Complex, waste management objectives and operations, and safety analysis. Extensive data concerning radioactive waste content and activity, effects of array size on criticality, and comparison of containers, are provided. (LKH)

BURIAL; COST BENEFIT ANALYSIS; CONTAINER INTEGRITY; CRITICALITY; DRAINAGE CHARACTERISTICS; EARTHQUAKES; REVIEWS; MONITORING; SAFETY; STORAGE, ABOVEGROUND; STORAGE, GEOLOGIC; TORNADOES; WASTE MANAGEMENT; WASTES, RADIOACTIVE

<177>
Kuhn, K., E. Albrecht, H. Kolditz, K. Thielemann, W. Diefenbacher, W.J. Engelmann, W. Krause, H.C. Schuchardt, and E. Smailos, Institut fur Tieflagerung, Gesellschaft fur Strahlen- und Umweltforschung mbH, Munich, German Federal Republic; Abteilung Behandlung Radioaktiver Abfalle, Gesellschaft fur Kernforschung, mbH, Karlsruhe, German Federal Republic.

Recent Results and Developments on the Disposal of Radioactive Wastes in the Asse Salt Mine.

CONF-760310; STI/PUB/433; IAEA-SM-207/82; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, (pp. 287-298), 828 pp.(1976)

Disposal of radioactive wastes into rock salt formations has been pursued in the Federal Republic of Germany since 1963. The Asse Salt Mine serves as the national R&D facility. 6,,000 containers with low-level wastes were disposed of from April 1967 till February 1976. In addition, 746 drums containing intermediate-level wastes were disposed of from August 1972 to February 1976. An R&D program has been formulated for test disposal of high-level wastes in Asse in the early 1980's. Another program, involving the use of a solution cavity in a salt dome for waste disposal, has been adapted to the Asse Mine. A man-made cavity is under construction as a substitute for a solutional cavity; disposal operations are expected to begin in early 1979. The Asse prototype cavity will test the possibility of direct disposal of intermediate-level wastes without shielding through a small diameter shaft into a cavity. (Auth) (LKH)

STORAGE, GEOLOGIC; WASTE STORAGE; WASTE DISPOSAL; WASTES, LOW-LEVEL; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, RADIOACTIVE; REPOSITORY; SALT DEPOSITS; SALT DOMES; MINES; CAVITIES

<178>
Lacy, W.J., Oak Ridge National Laboratory, Health Physics Division, Oak Ridge, TN.

Radioactive Waste Disposal Report on Seepage Pit Liquid Waste-Shale Column Experiment.

ORNL-2415; 21 pp.(1957, November 1)

While the Conasauga shale appears to have the capacity to adsorb liquid radwaste, questions of interest still remain: 1) Is there any way of telling when breakthrough of large quantities of radionuclides will occur; 2) what will be the breakthrough sequence for the radionuclides present in the waste; and 3) when the pits are abandoned, will the leaching action of rain and ground water remove dangerous quantities of activity from the saturated shale? Using an ion exchange technique about 10% of Ru 106 in a carbonate solution of pH 12 was removed on a cationic resin, 50% on an anionic resin, while 40% was not exchanged. The shale has an ion exchange capacity of 27.8 meq/100 g. Hence, about 1 million gallons of waste can be decontaminated adequately with 10 cu yd of shale and the shale will remove 85% of the total activity. Of the remaining 15%, most will be Ru 106. The 1 million gallons is not the total capacity of the shale but the breakthrough capacity. Ruthenium is the first radionuclide to breakthrough, followed by trivalent rare earths, then strontium and cesium. Capacity of the shale for cesium is 7.7 X 10(E+8) dpm/g. Upon leaching only 1% of the total activity in the column was found in the effluent. A material balance is also included. More work needs to be done on the breakthrough sequence, identification of the other radionuclides which are present in the waste solution, investigation of the change in concentration of ammonia before breakthrough, and more quantitative study on the affects of leaching a saturated column with tap water and with weak acid, salt solutions. (WDY)

pH; Ion Exchange Capacity

Ru 106; Cs 137; Sr 90; Rare Earths

ADSORPTION; ACIDS; ANIONS; BASES; BREAKTHROUGH DISTRIBUTION; CLAYS; CONTAINERS; EQUATIONS; FISSION PRODUCTS; LABORATORY STUDIES; LATTICE; MATERIAL BALANCE; RADIONUCLIDE MIGRATION; SEEPAGE PITS; SHALES; SOILS; WASTE DISPOSAL; WASTES, LIQUID; WASTES, RADIOACTIVE; RESINS; CATIONS

<179>
Linderoth, C.E., and D.W. Pearce, Hanford Laboratories Operation, Richland, WA.

Operating Practices and Experiences at Hanford.

Ground Disposal of Radioactive Wastes, W.J. Hanhan (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 7-16), 160 pp. (1961, July)

Some of the operating practices and experiences at Hanford are presented. Waste solutions are disposed of to the ground in naturally-occurring depressions, known as swamps. The largest volume of waste water comes from the cooling water streams and to mid 1959 a total of 139 million liters containing about 2,500 curies have been discharged to the swamps. Trenches are used at Hanford for wastes with a high salt or complexed radioisotope content. Before such waste is placed in the trenches it is put in underground storage tanks. The depleted uranium is recovered and Sr 90 and Cs 137 are scavenged and returned to the tanks as sludge. The solution placed in the trenches contains Co 60 and at the time of the report 106 million liters of waste containing 647,000 Ci of fission products had been discharged. Process condensate wastes containing 1 x 10(E-1) to 1 x 10(E-3) uCi beta emitters/cc are disposed of in sub-surface cribs. These wastes are allowed to percolate through the soils to be held by ion exchange. A total of 72 cribs have received 18.9 billion liters with 7.9 million Ci of fission products. It is estimated that 75% of the "gross beta-emitters" discharged have decayed to non-radioactive nuclides. Process engineering has reduced the amount of radwaste discharged each month from 18,000 to 180 Ci without changing the waste volume discharged. A well study at one of the cribs indicated that Cs 137 was retained in the soils immediately below the crib, Sr 90 was contained on the upper soils, and Ru was widely spread over the area. At present no demonstrable hazards have been created by the disposal of wastes to the ground at Hanford. (NDV)

Diagrams indicating the spread of the radionuclides are included.

Waste Volume

Sr 90; Cs 137; Co 60

BETA PARTICLES; CONCENTRATION; CONTAINMENT; CONTAMINATION; DISPOSAL SITE; ENVIRONMENT; FIELD STUDIES; FISSION PRODUCTS; ION EXCHANGE; MONITORING; RADIONUCLIDE MIGRATION; SLUDGES; SOILS; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, LIQUID; WASTES, INTERMEDIATE-LEVEL; WASTE VOLUME

<180>
Marine, I.W., Savannah River Laboratory, Aiken, SC.

Geochemistry of Ground Water at the Savannah River Plant.

DP-1356; 102 pp. (1976, September)

Subsurface hydrogeologic systems underlying the Savannah River Plant (SRP) were studied to determine the origin and age of the contained fluids. Three distinct systems exist beneath SRP: the Coastal Plain sediments, crystalline metamorphic basement rock, and a Triassic rock basin surrounded by the crystalline rock. The water in the Coastal Plain sediments is low in dissolved solids (about 30 mg/l), acidic (pH about 5.5), and comparatively recent. Water in the crystalline rock is high in dissolved solids (about 6000 mg/l), alkaline (pH about 8), and about 800,000 years old as determined by helium dating techniques. Water in the Triassic rock is highest in dissolved solids (about 18,000 mg/l) and is probably older than the water in the surrounding crystalline rock; a quantitative age was not determined. The origin of the water in the crystalline and Triassic rock could not be determined with certainty; however, it is not relic sea water. A detailed geologic-hydrologic history of the SRP region is presented. (Auth)

Data are presented in tables at the back of the report. A good summary of the geologic-hydrologic history is also included in the tables.

Stratigraphic Unit Thickness; pH

H 3; O 18; S 34

ALKALINITY; CALCIUM COMPOUNDS; CHLORINE; CHEMICAL ANALYSIS; DRAINAGE; GEOCHEMISTRY; GROUND WATER; HELIUM; HYDROLOGY; LEACHING; METAMORPHIC ROCKS; OXIDATION; pH; ROCKS; WATER TABLE; SEDIMENTS; GEOLOGY; BEDROCK

<181>
Markham, O.D., K.W. Puphal, and T.D. Filer, Idaho National Engineering Laboratory, Idaho Falls, ID; Los Alamos Scientific Laboratory, Los Alamos, NM.

Plutonium and Americium Contamination near a Transuranic Storage Area in Southeastern Idaho.

Journal of Environmental Quality 7(3):422-428. (1978)

From 1954 to 1970, 48,000 Ci of Am 241, 21,000 Ci of Pu 239, and 570 Ci of Pu 238 were buried at the Subsurface Disposal Area(SDA) at the INEL Radioactive Waste Management Complex. The transuranic isotopes are contained in about 65,200 cu m of solid waste. Highest concentrations of Pu 238 (0.8 pCi/g), Pu 239 (36.5 pCi/g), and Am 241 (51 pCi/g) were found in the surface soils (0-4 cm) on the perimeter area where water drains from the SDA. At the sampling station in the northeast corner the highest levels were found in the deeper soils (4-8 cm). The general contamination patterns for all three isotopes were spreading in NE-SW directions. For all three nuclides in the 4 to 8 cm depth concentrations were lower than those detected in the 0 to 4 cm depth. A paired T test indicated no significant difference in Am 241/Pu 239 ratios for the two soil depths. But the Pu 239/Pu 238 ratios in the 0 to 4 cm depth were significantly higher than the ratio in the 4 to 8 cm soil depth. These data indicate that Pu 238 vertical migration into the 4 to 8 cm soil depth is probably occurring faster than migration of Am 241 or Pu 239. Drainage topography suggests that the relatively high concentrations of Pu and Am in the soils in the perimeter drainage depression near the NE corner resulted when water from melting snow near the SDA flooded the SDA in 1962 and 1969. Although the floods are responsible for the higher

<181>

DISPOSAL SITE

<181> CONT.
concentrations on the perimeter there is no evidence to indicate water is the primary transport mechanism for contamination outside the perimeter. Hides and gastrointestinal tracts of deer mice had higher concentrations of transuranics than lungs or carcass. Ingestion appeared to be a more important mechanism than inhalation. (NDV)

Waste Volume

Am 241; Pu 239; Pu 239

AMERICIUM; PLUTONIUM; WASTES, TRANSURANIC; DISPOSAL SITE; SOILS; RADIONUCLIDE MIGRATION; DRAINAGE; TOPOGRAPHY; WATER; BIOTA; CONTAMINATION; SAMPLES; STATISTICS

<182>
Maslovskii, R.Y.

Autonomous Aspiration Unit for Sampling Radioactive Aerosols from Air.

CONL/tr-8523; 3 pp.; Gigiena i Sanitariya 10:70-71.(1974)

A special aspiration unit was mounted in the cabin of a vehicle transporting radioactive material so monitoring could be done in the air during the collection, transportation and unloading at the burial site. The unit consists of a filter holder, a rotameter and a "Buran" suction pump connected to a 12 volt battery for operation. The battery in the vehicle may be used when necessary. The capacity of the unit is 300 liters/min and the weight of the unit is 8 kg. This unit makes continuous operative monitoring of the level of air contamination by radioactive materials possible. (NDV)

AEROSOLS; CONTAMINATION; FIELD STUDIES; FILTERS; MONITORING; POLLUTION, AIR; WASTES, RADIOACTIVE; DISPOSAL SITE; TRANSPORTATION; EQUIPMENT; METHODS

<183>
McHenry, J.R., Hanford Atomic Products Operation, Richland, WA.

Fixation of Radioisotopes on Hanford Soil Types.

TID-7550; Fixation of Radioactivity in Stable, Solid Media, Proceedings of a Conference, Baltimore, Maryland, June 19-21, 1957. U.S. Atomic Energy Commission, Oak Ridge, TN, (pp. 30-35), 102 pp.(1958, March)

Information obtained at Hanford concerning the fixation of radioisotopes on soils was derived from experiments utilizing composite samples of subsoils typical of the disposal area of interest. These composites are calcareous sands containing 1.5 to 8% clay and about 1.5% calcium carbonate. The subsoils consist principally of unconsolidated, relatively unweathered sediments made up of lake and river deposited gravels, sands, silts, and clays. The pH of the soil paste is about 8.0. The clay fraction consists largely of montmorillonite clay and calcite with a small fraction of chlorite. The composite samples have a total cation exchange capacity of about 6 meq/100 g of soil. The total exchange capacities of the composite samples are largely due to the contributions of the clay fraction. The cation exchange capacities of some subfractions of the clay fraction are 90 to 115 meq/100 g for the less than 0.2 micron fraction; 60 to 100 meq/100 g for the 0.2 to 1.0 micron fraction; and 50 to 80 meq/100 g for the 1.0 to 2.0 micron fraction; and 50 to 80 meq/100 g for the 1.0 to 2.0 micron fraction. The adsorption by soil of the studied isotopes from solutions whose concentrations varied from 10(E-3) to 10(E-8) moles/liter, and less is attributed to ionic exchange. The equilibrium adsorption of radioisotopes from solutions containing less than 10(E-5) moles/liter of the isotope is, in decreasing order: Pu-Ce-Y-Cs-Sr. Average Kd values of these isotopes in Hanford type soil low-salt waste (distilled waste) systems are: Pu, 1980; Ce, 300; Y, 150; Cs, 125; and Sr, 70. The Kd values for each of these isotopes remains constant with changes in the isotopic concentration as long as the concentration of the cation of concern is less than 0.1% of the saturation capacity of the soil. If a sample of the test soil is treated with Cs 137 or Sr 90 until 0.1% of the saturation capacity of the soil is occupied by either cation the soil will contain some 1.1 mCi of Cs 137/g or 0.5 mCi of Sr 90/g, provided the solution contains no competing ions. The adsorption of radioisotopes from solution by ion exchange on soils is an equilibrium reaction and so the removal of the exchanged radioisotopes can be effected by varying the conditions of the system. Cesium, strontium, cerium, yttrium, and plutonium are all readily and completely removed when the test soil is treated with 0.1 to 1.0 N HCl. Various salt solutions will accomplish the same result, but require larger volumes. (JMT)

Distribution Coefficient; Ion Exchange Capacity; pH; Clay Content; Calcium Carbonate Content; Saturation Capacity

Cs 137; Sr 90; Pu; Y; Ce

MONTMORILLONITE; CALCITE; SOILS; CLAYS; SILTS; pH; ION EXCHANGE; ION EXCHANGE CAPACITY; ADSORPTION; CATIONS; DISTRIBUTION COEFFICIENT; SALTS; ISOTOPES; WATER; SATURATION; SEDIMENTS; SANDS; GRAVELS; LABORATORY STUDIES

<184>
McKinley, K.B., and J.D. McKinney, EG and G Idaho, Incorporated, Idaho Falls, ID.

Retrieval of Buried Transuranic Waste at the Idaho National Engineering Laboratory.

CONF-780622; American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (p. 364); Transactions of the American Nuclear Society 28:364.(1978, June)

Two projects have been established at the INEL Radioactive Waste Management Complex for the retrieval of approximately 65,000 cu m of TRU wastes buried there between 1954 and 1970. The Initial Drum Retrieval (IDR) Project has safely recovered, repackaged, and placed into interim storage 18,000 drums of TRU waste buried from 8 to 10 years. IDR operations were carried out under a 52.5 x 36 x 12.2 m air-support weather shield (ASWS) using a Drott excavator. Removal was at a rate of about 100 drums/day. Intact drums were placed in cargo containers and sealed with an elastomeric sealant. (Leached or contaminated drums, (7%), were wrapped in

<184> COST.
plastic, placed in DOT 7A metal boxes, and interim-stored above ground. The Early Waste Retrieval (EWR) Project is developing and investigating techniques for retrieval of waste buried up to 28 years. It is carried out under an ASWS, within an Operating Area Confinement (OAC) building under negative pressure. Waste is removed by backhoe, wrapped in plastic, placed in metal boxes, and stored above ground. Operations began with wastes buried 15-17 years. To date, 132.3 cu m of TRU waste has been recovered or generated, including 320 drums (67.7 cu m of waste), 33.3 cu m of uncontained waste, 19.4 cu m of contaminated soil, and 12.9 cu m of operations-generated waste. About 70% of the drums were breached, of which 9% leaked free liquid and 5% had external contamination. Alpha levels above 9,000,000 dis/min have been frequently encountered in loose waste and contaminated soil. Results of both projects indicate that on a small scale, safe retrieval and handling of buried TRU waste is possible. They have provided the foundation for possible future retrieval efforts at the RWMC. (Auth) (LBM)

Waste Volume

WASTE MANAGEMENT; RETRIEVABILITY; WASTE STORAGE; METHODS; CONTAINERS; BOXES; STORAGE, ABOVEGROUND; WASTES, RADIOACTIVE; WASTES, TRANSURANIC; FIELD STUDIES; DRUMS

<185>
McKinley, K.B., and J.D. McKinney, EG and G Idaho, Incorporated, Idaho Falls, ID.

Initial Drum Retrieval Final Report.

TREE-1286; 23 pp.(1978, August)

The Initial Drum Retrieval (IDR) program was initiated in FY-1978 to retrieve drums buried between 1968 and 1970. Retrieval operations began in July 1978, and concluded in June 1979. During this period, a total of 20,262 drums with a TRU waste volume of 8391 cu m and 1 cu m of contaminated soil were retrieved. Of this total, 18,029 drums plus the contaminated soil were repackaged and stored on 20-year interim storage pads. One drum containing Co 60 and Cs 137 sources was disposed into Pit 15. Evaporated salts, contained in 2232 drums, were disposed in the Transuranic Disposal Area. The condition of the buried drums limited IDR retrieval operations to Pits 11 and 12. The retrieval operations in Pits 11 and 12 were conducted under an Air Support Weather Shield (ASWS) for weather protection. A Drott excavator was used for retrieving the drums. After retrieval techniques were optimized, an average of 750 drums per month were retrieved. The retrieval operations were completed without serious injury or spread of contamination into the environment. The cost to retrieve 20,262 drums was $1,618,820 or about $80 per drum. The average retrieval cost per drum during the last 2 years of operation was about $60. (Auth)

DRUMS; BURIAL; RETRIEVABILITY; WASTES, TRANSURANIC; ECONOMICS; WASTE STORAGE; WASTE MANAGEMENT

<186>
McKinley, K.B., and J.D. McKinney, EG and G Idaho, Incorporated, Idaho Falls, ID.

Early Waste Retrieval Interim Report.

TREE-1265; 31 pp.(1978, May)

The Early Waste Retrieval Project began in 1976 to investigate methods, risks, and hazards associated with retrieval from subsurface burial of 65,000 cu m (2,300,000 cu ft) of waste contaminated with transuranic isotopes. The retrieval facilities include an Air Support Weather Shield and a moveable Operating Area Confinement. During this reporting period (October 1976 through September 1977), waste retrieved from Pit 2 and Trenches 8 and 10 consisted of loose waste, drums, and contaminated soil. Of the retrieved drums, 70% from Pit 2 and Trench 10 and all from Trench 8 were severely breached. Free liquids leaked from about 9% of the drums, and 5% were externally contaminated. Although alpha-contamination levels greater than 2 X 10(6) cpm were frequently encountered, available equipment and established safety and operating procedures protected the personnel and prevented any contamination from being released into the environment. Other studies supported or conducted by the Early Retrieval Project are a transuranic isotope migration study, a container development study, and soil binder studies for dust and contamination control. Retrieval efforts showed that two trenches were about 6 m north of their present monuments. Also, two pits are about 6 m narrower than their monuments indicate. The total cost of the project during FY 1977 was $883,701. (Auth) (PAP)

WASTES, TRANSURANIC; BURIAL; RETRIEVABILITY; PITS; SOILS; DRUMS; TRENCHES; CONTAINERS; WASTE VOLUME; EQUIPMENT; PACKAGING; METHODS; RADIONUCLIDE MIGRATION; LEAKS; BINDERS; HAZARD ANALYSIS; SHIELDING; WINDS; ALPHA PARTICLES; PERSONNEL INSTRUMENTATION; HEALTH PHYSICS; SAFETY; EXCAVATION; COSTS

<187>
Metzger, J.C., and A.E. Snyder, Idaho National Engineering Laboratory, Idaho Falls, ID.

Transport, Handling, and Interim Storage of Intermediate-Level Transuranic Waste at the Idaho National Engineering Laboratory.

TREE-1172; 44 pp.(1977, September)

The transport, handling, and interim storage of intermediate-level transuranic (TRU) waste at the Idaho National Engineering Laboratory (INEL) is discussed.
Transuranium-contaminated solid waste is considered as such when the alpha-emitting radionuclides of long half-life and high specific radiotoxicity are greater than 10 nCi/g. Waste form criteria and packaging criteria are outlined for the Waste Isolation Pilot Plant and Radioactive Waste Management Complex. The current intermediate-level TRU storage facility contains 26 storage vaults. Twelve of these vaults have an inside diameter of 0.50 m and the remaining have an inside diameter of 0.39 m. The storage vaults are carbon steel pipe implanted vertically in a long, narrow berm with a crushed gravel base course and a subbase of pit run gravel. Each vault consists of a long, straight-walled storage section, a

<187>

DISPOSAL SITE

<187> CONT.
concentric reducer section, and large diameter, straight-walled lead-in section. A lid attached to the vault body seals the vault and any moisture is prevented by a bag of desiccant between the lid and the shield plug. Additional storage vaults can be accomodated at the facility. These newer vaults have been redesigned to better meet the Waste Isolation Pilot Plant criteria. Environmental monitoring is also done. The total cost of storage at the facility is $6921/cu m or $196/cu ft. Waste generators as well as transportation and handling are included in the presentation. (NDV)

ALPHA PARTICLES; BURIAL; CONTAINMENT; ECONOMICS; ENVIRONMENT; PACKAGING; STORAGE, ABOVEGROUND; TRANSPORTATION; WASTE MANAGEMENT; WASTE STORAGE; WASTES, INTERMEDIATE-LEVEL; WASTES, SOLID; WASTES, TRANSURANIC

<188>
Miyake, Y., and K. Saruhashi, Geochemistry Research Association, Tokyo, Japan; Meteorological Research Institute, Tokyo, Japan.

A Critical Study on the International Atomic Energy Agency (IAEA) Definition of High-Level Radioactive Waste Unsuitable for Dumping at Sea.

Papers in Meteorology and Geophysics 27(3):75-80. (1976, September)

The definition of high-level waste unsuitable for sea disposal set forth by the London Conference on the Dumping of Wastes at Sea, 1972, and adopted by the IAEA, 1975, has been critically reviewed. Several difficulties have been found with the original report on which the definition is based. The horizontal eddy diffusion coefficient [10(E+4) sq cm/sec] was under-estimated, resulting in extremely reduced concentration ratios of Ra 226 and Pu 239 [on the order of 10(E-2) and 10(E-3), respectively]. The IAEA assumed a daily seafood intake of 6 g/day per person in their calculations; the present study assumes 100 g/day. The original report also assumed that the ratio of the maximum concentration of a given radionuclide above the pycnocline to that below it was 100, based on the eastern North Atlantic, while in most oceans the ratio is 1-10. The combination of these two factors gives a limiting environmental capacity as much as 5000 times that calculated by the present study. On the basis of these calculations a revised definition of high-level waste unsuitable for sea disposal would include any material with more than 10 Ci/ton beta-gamma active waste (excluding tritium), 1 Ci/ton Sr 90 plus Cs 137, 10(E+4) ci/ton tritium, or wastes containing any long-lived alpha active waste such as Ra 226 or Pu 239 at all. It is concluded that the IAEA definition of unsuitable waste should be carefully reexamined. (LKN)

SEA DISPOSAL; WASTES, HIGH-LEVEL; RADION; PLUTONIUM; FOODS; STRONTIUM; TRITIUM; CESIUM; DIFFUSION; MODELS; MODELS, MATHEMATICAL; OCEANOGRAPHY; RELEASE LIMITS

<189>
Newcomb, R.C., J.R. Strand, and F.J. Frank, U.S. Geological Survey.

Geology and Ground-Water Characteristics of the Hanford Reservation of the U.S. Atomic Energy Commission, Washington.

U.S. Geological Survey Professional Paper 717; 78 pp. (1972)

The geology and hydrology of the Hanford Reservation, Washington, is investigated, with particular emphasis on those aspects relating to disposal of radioactive waste to the geologic environment, general geologic conditions at the plant site, and ground water availability for industrial and domestic water supplies. The basalt of the Columbia River Group, the conglomerate of the Ringold Formation, and the glaciofluviatile and fluviatile deposits are the important water bearing units. Permeability and transmissivity of the Ringold conglomerate are 450 gal/day/sq ft, and 50,000 gal/day/ft respectively and for the glaciofluviatile and fluviatile deposits they are 46,000 and 1.3 million, respectively. For the Ringold Conglomerate the effective porosity is 11% and about 22% for the younger deposits. Up to 1959, the Hanford Reservation received 95% of the 2.6 million curies of radioactivity in low- and intermediate-level liquid waste disposed of to the ground in the U.S. Waste management methods is used at the plant and improvements over the past twenty years are discussed, including retention basins, pits and trenches, cribs and tile fields, burial grounds and vaults, tank farms, injection wells, swamp areas, and unplanned disposals (accidental spills). The tank and crib disposals contain the bulk of the radioactivity; these are placed above the water table in glaciofluviatile and fluviatile terraces in northwestern part of the reservation. Some activity is carried by ground water movement into the underlying Ringold Formation, and occasionally into the top part of the basaltic bedrock (Columbia River Group). Part of this activity reaches the large ground water "mounds" which are built up mainly by water from the waste swamps. Most radioactive cations are fixed fairly rapidly by earth materials; only Ru 106 and H 3 are generally free to migrate with the water. The potential hazard from tank- and crib-disposed wastes is discussed briefly. Controls on ground water movement by manipulation of the water table is considered. Well descriptions, logs, and water analyses, as well as geohydrologic maps are provided. (LKN)

Tables for the well data, stratigraphic description, and water analyses are included in the back. (RSIC/NDV)

Depth to Water Stratigraphic Unit
Thickness; Per. ty; Porosity; Transmissivity

AQUIFERS, ALLUVIUM; BASALTS; DRAINAGE; BASINS, DRAINAGE; DRAINAGE CHARACTERISTICS; FIELD STUDIES; GEOLOGY; HYDROLOGY; STRATIGRAPHY; GROUND WATER; LOGGING, WELL; MAPS; POLLUTION, WATER; SITE SURVEILLANCE; SOILS; SEDIMENTS; STORAGE, GEOLOGIC; TILL; TRENCHES; CRIBS; WASTE MANAGEMENT

<190>
Oertel, G.K., U.S. Department of Energy, Washington, DC.

Nuclear Waste Management.

<190> CONT.
Waste Management and Fuel Cycles '78, R.G. Post and W.R. Wacks (Eds.), Proceedings of a Symposium, Tucson, AZ, March 6-8, 1978. Arizona Board of Regents, (pp. 38-48), 661 pp. (1978, March)

A brief discussion on the various DOE programs for handling nuclear wastes is presented. One of the programs is to assure the management of nuclear wastes from their points of origin to their ultimate disposal in deep or shallow geologic formations or by other means. This includes all aspects of the technology and construction of the facilities. The commercial national waste terminal storage program has been redirected, as a result of President Carter's review of nuclear wastes produced in this country, to provide the capability for long-term storage and disposal of spent fuel in deep geologic formations. For defense high-level wastes an interim program was established that aims at volume reduction by solidification and storage in double shell tanks, and a long term program that will place minimal reliance on future monitoring. All TRU wastes will ultimately be placed in geologic storage. The low-level wastes will continue to be buried. The major elements of the DOE isolation program are the Waste Isolation Pilot Plant in salt (WIPP) and the National Waste Terminal Storage Program. (WDV)

BEDROCK; BURIAL; CONTAINMENT; DISPOSAL SITE; ENVIRONMENT; LEGISLATION; METHODS; REPOSITORY; SALTS; VOLUME REDUCTION; SOLIDIFICATION; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, TRANSURANIC

<191>
Pearce, D.W., C.E. Linderoth, J.L. Nelson, and I.L. Ames, Jr., General Electric Company, Hanford Atomic Products Operation, Richland, WA.

A Review of Radioactive Waste Disposal to the Ground at Hanford.

Disposal of Radioactive Wastes, Proceedings of a Conference, Monaco, November 16-21, 1959, (pp. 385-363). (1960)

The disposal of radioactive wastes to the ground from the Hanford separations plants is summarized; volumes of wastes and contained curies discharged to swamps, trenches, and cribs are presented. By June 1959, a total of 139 billion liters water containing 2,500 ci from the process cooling-water had been discharged. Significant literature on ion-exchange studies using Hanford soils is reviewed. The breakthrough of Sr 90 at varying concentrations and pH is discussed extensively. A field experiment with a model crib is described; preliminary findings indicate that spreading a waste solution below a disposal facility may be of greater benefit in disposal operations than previously assumed. Further studies with the calcite-phosphate reaction are reported; significant variables which affect Sr(+2) removal are calcite surface area, pH of the influent, flow rate, temperature and phosphate ion concentration. Bone-seeking radioisotopes are also removed from P-solutions when contact is made with calcite. The zeolite clinoptilolite shows high selectivity for Cs+ even in the presence of increased Na+ concentration. The effect of the large-scale ground disposal operation at Hanford is described by means of a map and geological cross-sections showing areas and volumes of ground and ground water contamination. Ruthenium, a major ground water contaminant, was found to be spread more widely than any other isotope. (Auth) (WDV)

Grain Size Distribution; pH; Waste Volume; Porosity; Distribution Coefficient

Sr 90; Cs 137; Ru 106; Co 60

ADSORPTION; BREAKTHROUGH DISTRIBUTION; BETA PARTICLES; BURIAL; CATIONS; CONTAMINATION; CONTAINMENT; DISPOSAL SITE; FIELD STUDIES; FISSION PRODUCTS; GEOLOGY; HYDROLOGY; ION EXCHANGE; POROSITY; PRECIPITATION, CHEMICAL; SOILS; WASTE DISPOSAL; WASTES, LIQUID; WASTES, RADIOACTIVE; WELLS

<192>
Peckham, A.E., U.S. Geological Survey, National Reactor Testing Station, Idaho Falls, ID.

Underground Waste Disposal Studies, Chemical Processing Plant Area.

Ground Disposal of Radioactive Wastes, W.J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 157-166), 168 pp. (1961, July)

Information collected during a series of well monitoring tests is presented. The Chemical Processing Plant (CPP) at the National Reactor Testing Station discharges 27 million gallons of liquid waste/month through a disposal well. The disposal well is about 600 feet deep, which is about 150 feet below the regional water table. The area is underlain by the Snake River basalt, which is overlain by sands and gravels up to 50 feet thick. These deposits are intermittent perched-stream sands and gravels and exhibit cross-bedding. Along with the sands and gravels are volcanic pillow lavas, cinders, or clinkers. As a result of the complex stratigraphy perched water beds exist. The water moves along the horizontal interflow zones and cinder beds with a hydraulic gradient of 3 to 5 feet per mile. For the Snake River basalt the transmissivity and storage coefficient are 3.3 X 10(E+6) gal/day/ft and 0.6, respectively. From several series of test wells saline water has spread in all directions from the disposal well. Ruthenium 106 was detected recirculating from the disposal well through the production wells and passing through the basalt aquifer. At a distance of 700 feet from the disposal well the saline water occurs over an area at least 1,200 feet wide. The pumping of water from production wells and disposal of water has created mounds and depressions in the water table. Furthermore, band or bands of saline water at one or more horizons are moving southeast and southwest. The saline water is being diluted. Whether this is a result of mixing or interfingering was not determined. A lag in time between change in plant discharge to the disposal well and changes in chloride concentrations indicate straight-line ground water movement rate of 15 to 50 ft/day. More tests are anticipated to further define the aquifer characteristics. (WDV)

Waste Volume; Depth to Water Table; Stratigraphic Unit Thickness; Hydraulic

<192>

DISPOSAL SITE

<192> CONT.
Gradient; Transmissivity; Storage Coefficient

to 106

AQUIFERS; BASALTS; CONTAINMENT; CONTAMINATION; DISPOSAL SITE; EFFLUENTS; GEOLOGIC FORMATIONS; GROUND WATER; HYDRODYNAMICS; MONITORING; PERCHED WATER; RADIONUCLIDE MIGRATION; STORAGE COEFFICIENT; PIEZOMETRIC SURFACE; SANDS; GRAVELS; TRANSMISSIVITY; WASTE DISPOSAL; WASTE VOLUME; WASTES, LIQUID; WASTES, RADIOACTIVE; WELLS, INJECTION

<193>
Price, W.H., and K.R. Fecht, Atlantic Richfield Hanford Company, Richland, WA.

Geology of the 241-AX Tank Farm.

ARH-LD-128; 13 pp. (1976, April)

This document is a summary of the general geologic features of the Hanford Reservation 241-AX Tank Farm accompanying geologic maps and cross-sections. The purpose of this investigation is to provide information for the evaluation of impacts of tank leaks. Data for the maps and sections are based in 600 samples from 23 dry and 4 water wells which have been sampled at one-to five-foot intervals. Samples have been classified by grain size, CaCO3 content, and visual inspection. The 241-AX Tank Farm sections include approximately 218 feet of unconsolidated Pleistocene glaciofluvial deposits and 108 feet of the middle (gravel and sand) unit of the Pliocene Ringold Formation. The upper and lower units of the Ringold are missing beneath 241-AX. The top of the Columbia River Basalt Group lies about 60 feet below the bottom of the prepared cross-sections. The water table at 241-AX is within the Ringold Formation, 223 feet below the base of the tanks. (LKH)

Seven geologic cross-sections are included in the report as separate plat.

Calcium Carbonate Content; Grain Size Distribution; Stratigraphic Unit Thickness; Depth to Water Table

TANKS; GEOLOGIC FORMATIONS; GEOLOGIC MAPPING; GEOLOGIC STRATA; SOILS; SEDIMENTS; DISPOSAL SITE; GROUND WATER; WATER TABLE; BASALTS; ROCKS; SITE EVALUATION

<194>
Price, W.H., and K.R. Fecht, Atlantic Richfield Hanford Company, Richland, WA.

Geology of the 241-B Tank Farm.

ARH-LD-129; 18 pp. (1976, April)

This document is a summary of the general geologic features of the Hanford Reservation 241-B Trade Farm accompanying geologic maps and cross-sections. The purpose of this investigation is to provide information for the evaluation of impacts of tank leaks. Data for the maps and sections are based on 650 samples from 27 dry and 3 water wells which have been sampled at one-to-five-feet intervals. Samples have been classified by grain size, calcium carbonate content, and visual inspection. The 241-B Tank Farm sections include approximately 182 feet of unconsolidated Pleistocene glaciofluvial deposits and 40 feet of Eolian Silt, also known as Early Palouse soil. Below these units, there is 26 feet of Pliocene-age Ringold Formation. This is the middle, gravel and sand unit of the Ringold; the upper and lower units are absent under 241-B. The top of the Columbia River Basalt Group is an erosional unconformity lying at an elevation of 366 feet above sea level. The water table at 241-B occurs 211 feet below the base of the tanks, at an elevation of 403 feet above sea level. Calcium carbonate content ranges from 0.6% in the middle Ringold to 20% in the Eolian Silt. (LKH)

Ten geologic cross-sections are included in the report as separate plates.

Calcium Carbonate Content; Grain Size; Distribution; Stratigraphic Unit Thickness; Depth to Water Table

TANKS; GEOLOGIC FORMATIONS; GEOLOGIC MAPPING; GEOLOGIC STRATA; SOILS; SEDIMENTS; DISPOSAL SITE; GROUND WATER; WATER TABLE; BASALTS; ROCKS; SITE EVALUATION

<195>
Price, W.H., and K.R. Fecht, Atlantic Richfield Hanford Company, Richland, WA.

Geology of the 241-BX Tank Farm.

ARH-LD-130; 18 pp. (1976, April)

This document is a summary of the geologic features of the Hanford Reservation 241-BX Tank Farm with accompanying geologic maps and cross-sections. The purpose of this investigation is to provide information for the evaluation of impacts of tank leaks. Data for the maps and sections are based on approximately 800 samples from 32 dry and 4 water wells which have been sampled at one- to five-foot intervals. Samples have been classified by grain size, CaCO3 content, and visual inspection. The 241-BX Tank Farm sections include 188 feet of unconsolidated Pleistocene glace of burial deposits overlain by another 42 feet of the same, used as backfill for the waste storage tanks. Beneath the Pleistocene sediments are 48 feet of the middle unit of the Pliocene-age Ringold Formation. In the 241-BX area, this consists of coarse to medium sand with some silt and pebbles. The upper and lower units of the Ringold are missing beneath 241-BX. Bedrock is the Columbia River Basalt Group (Miocene-Pliocene), which contacts the Ringold unconformably at an elevation of 378 feet above sea level. The water table lies within the Ringold Formation at 403 feet above sea level, 211 feet below the base of the tanks. (LKH)

Ten geologic cross-sections are included in the report as separate plates.

Calcium Carbonate Content; Grain Size Distribution; Stratigraphic Unit Thickness; Depth to Water Table

TANKS; GEOLOGIC FORMATIONS; GEOLOGIC MAPPING; GEOLOGIC STRATA; SOILS; SEDIMENTS; DISPOSAL SITE; GROUND WATER; WATER TABLE; BASALTS; ROCKS; IGNEOUS ROCKS; SEDIMENTARY ROCKS; MAPS; FIELD STUDIES

<196>
Fortman, W.D., R.L. Wheeler, and R.A. Rogers, Los Alamos Scientific Laboratory, Los Alamos, NM.

Geologic Description of Cores from Holes P-3 HH-1 through P-3 HH-5, Area G, Technical Area 54.

LA-7309-MS; 13 pp.(1978, May)

Five horizontal holes were cored beneath Pit 3 near the southeast edge of Mesita del Buey at Area G. Cores were obtained at depths ranging from 0 to 7.2 m beneath the pit, from two units of a rhyolite tuff. Five horizontal holes were cored beneath Pit 3 near the southeast edge of Mesita del Buey at Area G. The pit, filled and covered by 1966, contains solid radioactive wastes. The holes were cored to obtain samples of the tuff underlying the pit to determine if there has been any migration of radionuclides by infiltration of water in the past 10 yr. The five holes were collared in Unit 2b or the Tshirege Member of the Bandelier Tuff; three of the holes plunged downward into Unit 2a. This report describes the rock units penetrated by core holes and the joint characteristics observed. The locations of core samples selected for analysis are related to the floor of the pit. Core recovery from the five holes ranged from 52% to 75%, with joint frequencies ranging from 0.9 to 1.7 m of core recovered. The largest percentage of the joints were filled or plated with brown clay. The remainder were open with joint faces slightly weathered or plated with caliche. Joint orientations on the mesa and in the pit indicate that the joints formed as the ashflows cooled. (PAF)

CORES; PITS; WASTES, SOLID; BOREHOLES; SAMPLES; TUFFS; WATER; RADIONUCLIDE MIGRATION; ROCKS; JOINTS; ROCK MECHANICS; CLAYS; CALICHE; ASHFLOWS; SITE SURVEILLANCE

<197>
Routson, R.C., Battelle-Pacific Northwest Laboratories, Richland, WA.

A Review of Studies on Soil-Waste Relationships on the Hanford Reservation from 1946 to 1967.

BNWL-1660; 59 pp.(1973, March)

Large volumes of waste liquids were disposed of in the sediments on the Hanford Reservation during the more than twenty years of operation. The low-level wastes, 5 x 10(E-5) to 100 uC/l, were placed in underground structures (cribs) to percolate through the sediment. The high-level (greater than 100 uC/l) were stored in underground tanks. The sediments in the Hanford area are thick glaciofluviatile, thin "Palouse Soil" and the Ringold Formation. The clay in these sediments contained montmorillonite, chlorite, and mica. The clay from the Ringold also had a fraction of kaolinite. Three zones of disposal were delineated; the vadose zone above 20 feet depth (unsaturated); the vadose zone below 20 feet (lower vadose zone); and the saturated zone. The lower vadose zone was the most desirable for waste disposal because of minimal chance of organic contact. The second most desirable zone was the saturated zone. Radionuclides normally found in the waste were Sr 90, Cs 137, Pu, Co 60, Ru 106, transition metals, rare earths, and H 3. For most of the study the maximum permissible concentration (MCP) of Sr 90 was considered to be the limiting factor. Since the lower vadose zone is where most of the long-lived radionuclides are stored, a large research effort at Hanford has been devoted to studying radionuclide retention. The three categories of research are: 1) attain maximum retention of radioactive wastes; 2) predict breakthrough of radioactive wastes into the groundwater; and 3) evaluate possible removal mechanisms of previously sorbed radionuclides. Under maximum retention ion exchange (particularly with clays), precipitation (rare earths' and transition metals' hydroxides) and replacement are considered. Also experimental methods for sorption measurement are discussed. Prediction of breakthrough is considered empirically as well as theoretically. Possible removal mechanisms are leaching, diffusion and particulate transport. For the saturated zone the increase in water table elevation and attendant changes in the flow system, introduction of radioactive contamination into the groundwater, and introduction of nonradioactive contaminants are considered. Some consideration is given to the upper vadose zone. (WDV)

H 3; Sr 90; Cs 137; Co 60; Pu; Ru 106

ABSORPTION; ALLUVIUM; ALPHA PARTICLES; ANIMALS; ANIONS; AQUIFERS, CONFINED; ATMOSPHERE; BETA PARTICLES; BREAKTHROUGH DISTRIBUTION; CATIONS; COMPUTER PROGRAMS; CONTAINMENT; DECONTAMINATION; DISPOSAL SITE; EQUATIONS; EXHAUST GASES; GROUND WATER; HYDROLOGY; INHALATION; ION EXCHANGE CAPACITY; KAOLINITE; LABORATORY STUDIES; LEACHING; LITHOLOGY; MAXIMUM PERMISSIBLE CONCENTRATION; MODELS, MATHEMATICAL; MONTMORILLONITE; pH; PREDICTIONS; HALF-LIFE, RADIOLOGICAL; RARE EARTHS; RETENTION PONDS; RIVERS; SEDIMENTS; CLAYS; SURFACE CONTAMINATION; TANKS; THEORETICAL STUDIES; UNSATURATED ZONE; WASTE DISPOSAL; WASTE STORAGE; WASTE WATER RESERVOIRS; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, TRANSURANIC; WATER TABLE; VEGETATION

<198>
Routson, R.C., W.H. Price, D.J. Brown, and K.R. Fecht, Rockwell Hanford Operations, Earth Sciences Group, Richland, WA.

High-Level Waste Leakage from the 241-T-106 Tank at Hanford.

RHO-ST-14; 247 pp.(1979, February)

The history, status, fate and impact of the 0.38 X 10(E+5) liter (115,000 gal) radioactive waste tank leak from the 241-T-106 tank have been studied since its discovery in 1973. This leak contained approximately 14,000 Ci of Sr 90, 40,000 Ci of Cs 137, and 8 Ci of Pu 239 and Pu 240; in addition, the leak contained approximately 270,000 Ci of radionuclides with half-lives of less than three years (99% of which were Ru 106). Thus, this leak was the largest single accidental release in the AEC's operational history. As of 1978, the maximum depth of the 1-uCi/liter Ru 106 concentration was found to be 31 meters (108 feet) below the ground surface in two wells; or, approximately 83% of the vertical distance from the bottom of the tank to ground water. In other wells, the vertical movement does not exceed 29 meters (95 feet). The maximum

<199>

DISPOSAL SITE

<198> CONT.
horizontal extent of the 1-uCi/liter Ru 106 concentration was found to be approximately 23 meters (75 feet) or about equivalent to the diameter of the tank. The rate of frontal movement of the leak was qualitatively estimated. A large portion of the movement occurred in 1973, prior to the publication of the initial status report by ARHCO. From 1973 to 1976, lateral movement occurred in at least some sediment layers (types) in all monitoring wells. From 1976 through 1977, waste movement could not be generally detected due to the radiodecay of mobile Ru 106 (half-life = 1 year), although some movement undoubtedly occurred. However, lateral movement was detected in three wells near the perimeter of the leak from 1977 to 1978. This movement probably resulted from the addition of water to the system down a well casing during an unusual climatic event during the winter of 1977-1978. The above general pattern of waste movement is about what would be expected by an infiltration of a pulse of water into a soil or sediment system. This generally consists of an initial rapid movement followed by an exponential decrease with time. Calculations on vadose zone moisture and wetting frontal movements were found to be essentially in qualitative agreement in their assessed lack of movement of any waste above concentration guidelines to the Hanford ground water. Thus, during the hazardous lifetime of the fission products, there will likely be no amount of radioactivity enter the Hanford ground water. Therefore, there appears to be no impact of the leak on the Columbia River (the nearest point of uncontrolled access). The 241-T-106 tank leak studies conducted up to the present time offer a sufficient basis of understanding of the 1973 leak. Thus, efforts will go from an active study and monitoring mode to a monitoring mode, exclusively. Since ungrouted wells can offer a path to deeper sediments of unexpected sources of water, all ungrouted wells in the T tank farm have been grouted and any new wells drilled in the T tank farm will be grouted as soon as the drilling is completed. In-well scintillation probe total gamma scans were run as soon as these wells were grouted to provide a basis for on-going monitoring. All T tank farm dry wells will continue to be monitored at least quarterly for the foreseeable future. (Auth) (LKM)

Ru 106

ACCIDENTS; TANKS; LEAKS; WASTES, LIQUID; WASTES, RADIOACTIVE; RADIONUCLIDE MIGRATION; MONITORING; INFILTRATION; SOILS; SEDIMENTS; WATER TABLE; VADOSE ZONE; WELLS; FIELD STUDIES; DISPOSAL SITE; WASTE DISPOSAL; WASTE STORAGE

<199>
Savannah River Laboratory, Aiken, SC.

Savannah River Laboratory Quarterly Report, Waste Management, July-September, 1975.

DPST-75-125-3; 80 pp. (1975)

The report provides current information on the Savannah River Laboratory development program for the management of radwastes produced by the Savannah River Plant. Sections on long-term waste management and interim management of plant waste comprise the report. Within the long-term management section, solid transuranic waste processing and storage, effects of heating concrete waste forms during curing and storage and effect of linear flow velocity on removal of Cs 137 by "Duolite" resin are discussed. The interim management of plant waste section considers waste tank corrosion, retention of organic-complexed plutonium in soil, metallurgical aspects of cleaning annuli of waste storage tanks, precise assay of high activity shipments of solid wastes, and fixation of cesium by soils. Topics abstracted separately are: effect of linear flow velocity on removal of Cs 137 by "Duolite" resin, retention of organic-complexed plutonium in soil, and fixation of cesium by soils. (NDV)

SORPTION; CONCRETES; CONTAINER INTEGRITY; CONTAMINATION; DECONTAMINATION; DISPOSAL SITE; FISSION PRODUCTS; GROUND WATER; LABORATORY STUDIES; MONITORING; FIXATION; FUEL REPROCESSING; SOILS; TANKS; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, SOLID; WASTES, RADIOACTIVE; WASTES, TRANSURANIC

<200>
Schultz, B.L.

Operating Practices, Experiences, and Problems at the National Reactor Testing Station.

Ground Disposal of Radioactive Wastes, W.J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 17-33), 168 pp. (1961, July)

The operating practices at the National Reactor Testing Station (NRTS) are outlined. NRTS is underlain by basalt of recent geologic age which is covered by sediments varying from a few inches to 100 ft in thickness. The soil is a silt loam and the subsoil consists of gravel and sand. Ground water occurs at depths ranging from 200 to 800 ft below the surface and the gradient is about 5 ft/mi. In general waste disposal at NRTS follows the philosophy of confinement and dispersal. Disposal by dispersal is accomplished by discharging the waste directly to the water table, whereas confinement is retention by the soils and regolith. During 1959 about 650 million gallons of water were discharged of which 70% was to the ground water through a well at the Idaho Chemical Processing Plant. The radionuclides put into the groundwater were I 131, Zr 95, Nb 95, and Ce 144. A well system for monitoring the effects of liquid discharge to the ground water was drilled. It was found that the water moves through the structural voids in the basalt and not through the interbeds of the soil. A special test was run using fluorescein dye. It showed the average groundwater velocity is 23 ft/day in the basalt. It also provided a basis for reconsidering the reliability of using 350 ft/day for radionuclide travel. (NDV)

Depth to Water Table; Hydraulic Gradient; Waste Volume

I 131; Zr 95; Nb 95; Ce 144, Sr 90; Pr 144

BASALTS; CONTAMINATION; CONTAINMENT; DISPOSAL SITE; FLUORESCEIN; GROUND WATER; HYDRODYNAMICS; HYDROLOGY; MONITORING; RADIONUCLIDE MIGRATION; SOILS; LOAM; WASTE DISPOSAL; WASTE VOLUME; WASTES, LIQUID; WASTES, LOW-LEVEL; WELLS

<201>
Singh, N., and J.S. Marshall, Savannah State College, Biology Department, Savannah, GA; Argonne National Laboratory, Radiological and Environmental Research Division, Argonne, IL.

A Preliminary Assessment of Plutonium 239, 240 Concentrations in a Stream near Argonne National Laboratory.

Health Physics 32(3):195-198.(1977, March)

Sawmill Creek at the Argonne National Laboratory (ANL) was used to investigate the behavior of Pu 239 in a stream environment. ANL introduces radioactive wastes into the stream through the sewage plant, research reactors, 60 in. cyclotron, fuel fabrication facility, and several other research related activities. The amount of effluent is 0.77 million gal/day, and the creek flow is 8.1 million gal/day. Fresh water samples were taken downstream and upstream of the discharge point. Samples of the algae, isopods, and fish as well as sediments were also taken on both sides of the discharge point. The filterable solids from the effluent water contained 99% of Pu 239 in contrast to 37% from the upstream water sample. The effluent water contained twice as much suspended matter (18.9 mg/l) as the upstream water (7.7%) and the downstream sediment had a higher Pu 239 concentration. The downstream fish had a higher Pu 239 activity in the G.I. tracts than in the gills. Plutonium activity in the insect sample was 78.0 plus or minus 27.0 fCi/kg wet weight. The downstream plutonium levels in the whole water were much higher than those reported in related studies. Apparently, Pu 239 is primarily scavenged by algae downstream. However food-chain discrimination against Pu 239 could not be demonstrated. (NDV)

Waste Volume; River Flow Rate

Pu 239

BIOTA; BIOSPHERE; CREEKS; CONTAMINATION; DISPOSAL SITE; FIELD STUDIES; FISH; MONITORING; PLUTONIUM; SEDIMENTS; WASTES, LIQUID; WASTES, RADIOACTIVE; WATER; WASTE VOLUME

<202>
Smailos, E., W. Diefenbacher, E. Korthaus, and W. Cooper, Gesellschaft fur Kernforschung mbH, Karlsruhe, German Federal Republic.

Radiolysis and Temperature Effects in Case of Underground Storage of Bitumen.

KFK-2329; 15 pp.(1976, May)

This study attempts to define limits to the specific activity of bitumenized wastes stored in the prototype 10,000 cu m Asse II cavity which will avoid buildup of combustible radiolytic gases and overheating of the wastes during storage. Maximum admissible temperature was set at 70 C, the softening point of bitumen Mexphalt 15. Calculations are presented showing that for fission product ages of 0.5, 1, and 2 yrs, limit values for specific activity are 0.6, 0.85, and 0.35 Ci/l, assuming instantaneous filling of the storage cavity, or 0.7, 0.85 and 0.3 Ci/l, assuming step-by-step filling. Calculations of combustible gas evolution, mainly H2, assume gas evolution of 4 vol% and filling of the cavity with waste at a rate of 500 cu m/yr over 5 and 10 yr periods. Specific activity limits for a 25% filled cavity are 0.70, 0.43, and 0.26 Ci/l for waste aged 0.5, 1, and 2 yrs, respectively; the limit values of absorbed dose until total decay for the same waste is 2.6 X 10(E+7) rads. Specific activity limits for a 50% filled cavity are 0.26, 0.18, and 0.09 Ci/l for waste aged 0.5, 1, and 2 yrs, respectively; the limit value of absorbed total dose until total decay is 0.6 X 10(E+6) rads. Safe disposal of wastes with activities up to 1 Ci/l can be facilitated by ventilation or introduction of a cover gas, by the construction of tall, narrow cavities for maximum heat dispersal, storage in media with high thermal conductivities, or by interim storage of wastes before permanent disposal. (LHN)

RADIOLYSIS; GASES; HYDROGEN; ORGANIC COMPOUNDS; BITUMENS; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; THEORETICAL STUDIES; SALT DEPOSITS; FIRE HAZARDS; HEAT SOURCES; WASTES, SOLID

<203>
Spitsyn, V.I., N.K. Pimenov, V.D. Balukova, Leontichuk, I.N. Kohorin, P.P. Yudin, and N.A. Rakov, USSR Atomic Energy State Committee, Moscow, USSR.

Foundational Prerequisites and Practice for Using Deep Water-Bearing Horizons for Disposal of Liquid Radioactive Wastes.

IAEA-CN-36/385; Nuclear Power and its Fuel Cycle, Proceedings of an International Conference, Salzburg, Austria, May 2-13, 1977. International Atomic Energy Agency, Vienna, (pp. 1-27). (1977)

A general discussion is presented on the methods and rational of storing wastes in geologic formations. When injecting or placing radwastes in such formations criteria must be set and adhered to so the radwaste does not reach the biosphere. Criteria such as low groundwater flow, structural and mineralogic integrity are considered. Research in treating high-level waste for geologic disposal includes washing the hole with weakly acid solutions and creating a transition zone between the stratum and wastes and introducing additional complexing agents into the wastes thus regulating the wastes. A test area is described for site suitability. (NDV)

AQUIFERS, CONFINED; COMPLEXES; CONTAINMENT; DISPOSAL SITE; GEOLOGIC FORMATIONS; GEOLOGY; GROUND WATER; HYDROLOGY; METHODS; RADIONUCLIDE MIGRATION; WASTE-ROCK INTERACTIONS; WASTE DISPOSAL; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LOW-LEVEL

<204>
Staples, B.A., G.S. Fornisk, and E.L. Wade.

Properties of Radioactive Calcine Retrieved from the Second Calcined Solids Storage Facility at Idaho Chemical Processing Plant.

ICP-1189; 55 pp.(1979, March)

The chemical and physical properties of radioactive alumina and zirconia calcine samples retrieved from the storage bins at

<204>

DISPOSAL SITE

<204> CONT.

the Idaho Chemical Processing Plant (ICPP) were measured. The data were compared with properties of samples taken during calcination of nonradioactive calcine. The calcine is representative of the two major types of high-level waste stored at ICPP: aluminum alloy fuel wastes and zirconium alloy fuel wastes. Chemical properties measured include chemical composition, crystalline structure, and radiochemical composition. The physical properties measured and reported include density, size distribution, relative attrition, solubility in 8M HNO3, thermal stability, and flow characteristics. The chemical and physical properties of the retrieved calcine after the 10-12 years of storage are very similar to freshly prepared simulated calcine. The results of this testing program indicate that the sample bins can be emptied pneumatically. (Auth) (NAP)

Density; Flow Rate

Sr 90; Co 60; Cs 134; Cs 137; Eu 154; Eu 155; Sb 125; Pu 238; Pu 239; Pu 240; Pu 241; Np 237; Am 241; U 234; U 235; U 236; U 238

RETRIEVABILITY; WASTE STORAGE; CALCINATION; SAMPLES; ALLOYS; ALUMINUM; ZIRCONIUM; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; RADIOCHEMISTRY; CHEMICAL COMPOSITION; DENSITY; PARTICLE SIZE; DISTRIBUTION; SOLUBILITY; THERMAL PROPERTIES; STRONTIUM 90; RADIATION, GAMMA; URANIUM; TRANSURANICS

<205>

Sumner, W.B., and L.L. Thomas, Allied-General Nuclear Services, Barnwell, SC.

Description of Allied-General Nuclear Services On-Site Solid Waste Storage Concepts.

Management of Low-Level Radioactive Waste, M.W. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 7, (pp. 679-698), 1218 pp. (1979)

This paper presents criteria and conceptual designs for interim storage of solid radioactive wastes at the Barnwell, South Carolina, reprocessing plant. Allied-General Nuclear Services (AGNS) will divide the majority of the contaminated solid waste generated during reprocessing of commercial spent nuclear reactor fuels into three categories: spent fuel cladding hulls, high-level general process trash (HLGPT), and low-level general process trash (LLGPT). The LLGPT will be stored in cargo containers identical to those used for road, rail, and sea transport. As these cargo containers are filled, they will be covered with earth for protection from natural phenomena. The cargo containers will be sufficiently monitored to allow detection and recovery of any radionuclides before they reach the environment. The hulls and HLGPT will be stored in caissons within separate engineered soil berms. The caissons will be lined and capped to provide sufficient protection from natural phenomena. The berms will include impervious clay layers at the bottom to prevent the downward movement of radionuclides and will be provided with sufficient monitoring to allow detection and recovery of radioactivity before it reaches the environment. The concept of utilizing engineered soil berms and soil-covered pads is believed to provide an optimum solution for short-term storage. Operating and capital costs must be balanced against requirements for flexibility and retrievability dictated by the uncertainties of future regulatory requirements for handling and disposing of these types of wastes. Despite the uncertainties and possibly short service life of these facilities, the design must still address current regulatory requirements related to the health and safety of the public. It is felt that the concepts presented meet these requirements while maintaining costs at a minimum. (Auth) (LKM)

DESIGN; STORAGE, INTERIM; WASTES, SOLID; PLANTS, FUEL REPROCESSING; CLADDING; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; CONTAINERS; CASKS; BERMS; CAISSONS; CAPPING; LINERS; CLAYS; MONITORING; STORAGE, ABOVEGROUND; COST BENEFIT ANALYSIS; REGULATIONS; WASTE MANAGEMENT; DISPOSAL SITE; HULLS

<206>

U.S. Department of Energy.

Alternatives for Long-Term Management of Defense Transuranic Waste at the Savannah River Plant, Aiken, South Carolina.

DOE/SR-WM-79-1. (1979, July)

Risks and costs have been calculated for 12 alternatives for the long-term management of defense transuranic waste now stored in trenches and on concrete pads at the Savannah River Plant. These alternatives are: (1) waste is left as is in trenches and on pads; (2) all waste is placed in onsite trenches; (3) all waste is placed in onsite trenches with improved cover; (4) retrievable waste is overpacked and placed in retrievable storage or in a permanent disposal facility (retrieval possible only during an initial testing period); (5) all waste exceeding 10 nCi/gm activity is retrieved, placed in storage or in a disposal facility; (6) retrievable waste is reduced in volume and placed in steel containers in storage or in a disposal facility; other waste is left in trenches; (7, 8) retrievable waste is reduced in volume, fixed in concrete (Alternative 7), or in glass (Alternative 8), and stored or disposed; other waste is left in trenches; (9) retrievable waste is processed as in (7, 8), but the other waste is left in trenches with improved cover; (10) all waste is retrieved, reduced in volume, and placed in steel containers for storage or disposal; (11, 12) all waste is retrieved, reduced in volume, fixed in concrete (Alternative 11), or in concrete and glass (Alternative 12), and stored or disposed. Risks and costs are estimated for a 110-year period, which allows a 10-year period for retrieving and processing the waste and a 100-year period for surveillance of storage or disposal methods. Risks are also estimated for an additional 1000 years, during which it is assumed that institutional control of the waste storage or disposal sites has ceased. Radiological risks are calculated for an individual at the fence line, and to the population residing in the most populated sector within 80 kilometers (50 miles) of the plant boundary. In addition, occupational exposure and non-radiological fatal injuries are estimated for the workers involved in constructing the facilities and retrieving and processing the waste. Risks to the

<206> CONT.
population for all alternatives are small compared with the radiation dose from natural radiation, while the estimated health effects (total cancers) are small compared with the number of observed fatal cancers. Among the 12 alternatives, costs vary by a factor of 20, from 28 million (1979) dollars to 512 million. Risks integrated for a 110-year period vary by a factor of 300, from 45 man-rems to 13,100 man-rems. Least cost and maximum risk occur for the no-action alternative (Alternative 7). The smallest risk occurs for alternatives in which all the waste is placed in trenches or in which the waste is overpacked in a short time period and placed in a deep geologic disposal facility. The maximum cost occurs for the alternative in which all of the waste is retrieved, fixed in glass or concrete, and disposed of in an onsite bedrock disposal facility. Risks for the longer 1100-year period vary by a factor of 10, from 200-370 man-rems for the alternatives in which all waste is processed (Alternatives 11 and 12) to 20,000 man-rems for the no-action alternative. (Auth) (LKH)

Extensive tabular data are provided in the text.

CONTAINERS; WASTE STORAGE; VOLUME REDUCTION; CAPPING; WASTES, RADIOACTIVE; STORAGE, ABOVEGROUND; OVERPACKING; STORAGE, GEOLOGIC; FIXATION; VITRIFICATION; CONCRETES; DISPOSAL SITE; COST BENEFIT ANALYSIS; WASTE MANAGEMENT; WASTE DISPOSAL; WASTES, TRANSURANIC; HAZARD ANALYSIS; RETRIEVABILITY; WASTE PROCESSING; WASTE TREATMENT; EXPOSURE, OCCUPATIONAL; EXPOSURE, POPULATION; RADIATION EFFECTS; COSTS; TRENCHES; BURIAL, SHALLOW; WASTES, DEFENSE

<207>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Present Environment of the Idaho National Engineering Laboratory (Including RWMC, Site 14, and Lemhi Range).

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 3, (pp. 3-1 - 3-36), 586 pp. (1979, February)

The environmental setting of the alternative disposal sites studied for long-term management of TRU waste stored at INEL is described in this section. The four disposal sites, RWMC (present location of the waste), Site 14, and the two sites in the Lemhi Range, are within the INEL boundaries. General descriptions of geographical setting, topography, climate and meteorology, geology and seismology, and hydrology are given in subsections. Also discussed are ecological aspects, archaeology, land use, population characteristics, and economical setting. Where there are local differences among the four locations with INEL, the differences are described. (RAF)

DISPOSAL SITE; GEOGRAPHY; WASTES, TRANSURANIC; SITE EVALUATION; TOPOGRAPHY; METEOROLOGY; GEOLOGY; SEISMOLOGY; HYDROLOGY; LAND USE; POPULATIONS; ECONOMICS

<208>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

The Radioactive Waste Management Complex and Its Operations.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 4, (pp. 4-1 - 4-16), 586 pp. (1979, February)

The Radioactive Waste Management Complex (RWMC) was established in the southwestern corner of the INEL in 1952 as a controlled area for burial of solid radioactive waste generated by INEL operations. It encompasses about 144 acres. Since 1954, waste contaminated with transuranic radionuclides has been received from defense-related activities at the Rocky Flats Defense Production Plant and from other operations of the Atomic Energy Commission and its successor agencies. Waste materials include laboratory equipment, process equipment, protective clothing, maintenance tools, and building materials. In terms of activity, the principal radionuclides in the TRU waste are Pu 238, 239, 240, 241 and 242, and Am 241. Approximately 2,200,000 cu ft of TRU waste was buried in pits and trenches at the RWMC between 1954 and 1970. Additionally, about 500,000 cu ft of beta/gamma waste was mingled with the buried TRU waste. Since 1970, about 1,100,000 cu ft of TRU has been received and stored retrievably on aboveground asphalt pads at the RWMC. By 1985, an additional 800,000 cu ft is expected to be received and stored. This will result in a total of about 2,000,000 cu ft of stored TRU waste on the pads and about 2,000,000 cu ft of buried TRU waste in the pits and trenches. Some of the containers for buried waste deteriorated and released low levels of contamination into the soil. (Auth) (RAF)

WASTES, SOLID; WASTES, TRANSURANIC; PLUTONIUM 238; PLUTONIUM 239; PLUTONIUM 240; PLUTONIUM 241; PLUTONIUM 242; AMERICIUM 241; WASTE VOLUME; BURIAL; PITS; TRENCHES; WASTE STORAGE; ASPHALT; SOILS; CONTAMINATION; CONTAINERS

<209>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Effects of Current Transuranic Waste Management on the Workers and on the Public.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 6, (pp. 6-1 - 6-8), 586 pp. (1979, March)

The work force for storage and disposal operations at the Radioactive Waste Management Complex (RWMC) ranges from 15 to 20 people. Radiation dosimetry exposure records are kept for each individual at the RWMC. Since the dosimetry records are for cumulative exposure, separate personnel records are not available for TRU-waste-only operations. Health physics monitoring indicates that most personnel exposure results from handling and disposal of non-transuranic waste. Radiation dose records from 1976 and 1977 indicate that the individual annual external dose for operating personnel ranged from 40 mrem to 1870 mrem. A preliminary estimate of exposure due to handling TRU waste alone was made using an

<209>

DISPOSAL SITE

<209> CONT.

average radiation of 1 mR/hr at 6 ft from most of the containers. The radiation level during emplacement would be on the order of 1 mR/hr. It was estimated that an individual would receive a dose of 10 to 20 mrem during a full work week. With 25 weeks of operation per year (at the current rate of TRU receipt), an individual would receive an annual dose of about 400 mrem. These estimated doses are well below the ERDAA-0528 standard of 5000 mrem/yr for wholebody dose. During normal handling and storage of TRU waste, there was no identifiable TRU contamination available for inhalation or ingestion. Radiation effects on the public from RWMC operations are indistinguishable from the effects due to natural background radiation. (NAP)

WASTES, TRANSURANIC; PERSONNEL; WASTE STORAGE; WASTE DISPOSAL; RADIATION DOSE; EXPOSURE, OCCUPATIONAL; EXPOSURE, EXTERNAL; EXPOSURE, INTERNAL; MONITORING; STANDARDS, FEDERAL

<210>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Alternatives Involving Leave-in-Place Management.

DOE/ET-0029; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 8, (pp. 8-1 - 8-10), 586 pp.(1979, March)

In Alternative 1, the TRU waste would be left in place as is. The present cover of plywood, polyvinyl sheeting, and 3 ft of soil over the waste would be maintained. The present environmental surveillance at the Radioactive Waste Management Complex (RWMC) would be continued, with improved procedures incorporated as they are developed. The current soil sampling program at the RWMC involves tests for detecting Pu at the surface and at depths of 6 to 12 in. Air monitors are presently installed at several locations around the perimeter of the RWMC to determine airborne concentrations of radionuclides. The lack of surface water near the RWMC restricts water sampling activities to collecting runoff from rainfall or melting snow. All these procedures would be continued for 100 yr, the assumed minimum duration of societal control over such activities. It was conservatively assumed that, after the 100-yr period, reliance would be placed entirely on the current waste confinement system (containers, soil covering) for isolation from man and protection of the environment. In Alternative 2, the TRU waste would be left in place, but engineered confinement would be added. Three concepts were studied as ways of providing this additional protection. In Concept 1, an additional 10-ft cover of compacted clay, plus a 3-ft cover of basalt riprap, would be built over the existing mounds on Transuranic Storage Area pads. Concept 2 would include the clay and riprap cover of Concept 1 plus increased isolation provided by pressure-grout sealing of the sediments beneath the asphalt pad. In Concept 3, the waste would be immobilized in place by injecting grout into the waste and into the sediment beneath the pad. For all three concepts of improved confinement, the present maintenance and surveillance at the RWMC would be continued for 100 yr, as in Alternative 1, with improved procedures incorporated as they are developed. (NAP)

WASTES, TRANSURANIC; WASTE STORAGE; WASTE MANAGEMENT; MONITORING; SAMPLING; SITE SURVEILLANCE; SOILS; PLUTONIUM; RUNOFF; PRECIPITATION, METEOROLOGICAL; CONTAINMENT; CONTAINERS; CLAYS; BASALTS; GROUTING; SEDIMENTS; SEAL MATERIALS

<211>
U.S. Department of Energy, Washington, DC.

Western New York Nuclear Service Center Study, Final Report for Public Comment.

TID-28905-1; 56 pp.(1978, November)

Recommendations for responsibility allocation, description of the Western New York Service Center (WNYSC), potential impacts, and estimated costs of options for the future of the Center are presented. From 1966 to 1972 about 640 MT of nuclear fuel were processed. Of that waste 390 MT were from AEC's production reactor and 100 MT were from commercial reactor fuel supplied under the baseload contract. The rest came from public utilities. In 1972 the plant was shutdown to make modifications and at that time new criteria were imposed for the tornado and seismic integrity plant. Existing facilities at the site are spent fuel receiving and storage, reprocessing plant, high-level liquid waste storage, NRC-licensed waste burial ground, and New York state-licensed waste burial grounds. The burial grounds contain 139,000 cu ft of spent fuel hardware and other solid wastes and 2.4 million cu ft of contaminated wastes. Recommendations provided include development of immobilization technology for the high-level liquid wastes and enough funding be provided such that short-term and long-term risks can be determined with confidence and minimized. Technical options are decontamination, decommissioning and waste disposal; continued nuclear use of the site; nonnuclear use of the site. Each of the options is analyzed in accordance with the existing facilities. Institutional and financial responsibilities are also outlined. (SUM)

Appendices include the authorizing legislation for solving the reprocessing center problem, and a glossary of important terms.

COST BENEFIT ANALYSIS; DECOMMISSIONING; DECONTAMINATION; HAZARD ANALYSIS; METHODS; IMMOBILIZATION; BURIAL; NUCLEAR FACILITIES; REPROCESSING; SAFETY; SPENT FUELS; RECOMMENDATIONS; WASTES, HIGH-LEVEL; WASTES, SOLID; WASTES, LIQUID

<212>
Wallskog, H.A., Rockwell Hanford Operations, Richland, WA.

System Design for Retrieval of Solidified High-Level Wastes at Hanford.

(1977)

A waste retrieval system has been conceptually designed as a step in the process toward demonstrating the capability

<212> CONT.
to retrieve the projected 36,000,000 gal of radioactive salt cake and sludge wastes from underground storage tanks at Hanford. This functionally complete, totally remotely operable system consists of a large mobile platform containing all the tools and equipment necessary to recover, remove, and package wastes for transfer to an on-site processing facility. Under current plans, the capability of this system will be demonstrated first with nonradioactive tests in a prototype test facility, and second by retrieving wastes from one or more of the Hanford waste tanks. (Auth) (RAP)

RETRIEVABILITY; WASTES, HIGH-LEVEL; SLUDGES; EQUIPMENT; PACKAGING; WASTE STORAGE; WASTES, SOLID; REMOTE HANDLING

<213>
Weeren, H.O., Oak Ridge National Laboratory, Oak Ridge, TN.

Waste Disposal by Shale Fracturing at Oak Ridge National Laboratory.

Management of Low-Level Radioactive Waste, M.W. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 7, (pp. 717-728), 1218 pp. (1979)

This paper discusses the operational experience, monitoring techniques, and some environmental considerations of radioactive waste disposal by shale fracturing as practiced at Oak Ridge National Laboratory (ORNL). This process is used at ORNL for the permanent disposal of locally generated intermediate-level waste solutions. In this process, the waste solution is mixed with a solids blend of cement and other additives; the resulting grout is injected into an impermeable shale formation at a depth of 700 to 1000 ft (200-300 m) - well below the level at which groundwater is encountered. The shale fracturing facility consists of the injection well, a network of observation wells for determination of the orientation of the grout sheet, a network of rock-cover monitoring wells for verification of the continued impermeability of the shale above the grout sheets, storage tanks for the waste solution, storage bin for the dry solids mix, a mixer, a surge tank, and injection pump and associated high-pressure piping. A standby injection pump is rented during injections to clear grout from the injection well and other high-pressure piping in the event of failure of the main injection pump. The existing facility was built in 1963; a series of demonstration injections were made in 1964 and 1965, after which the facility became operational. Since 1966, the facility has been used to inject 1.8 million gal (8.6 million l) of grout containing 550,000 Ci of intermediate-level waste. This waste is alkaline (about 1M in NaNO3) and has a specific activity of approximately 1 Ci/gal (0.26 Ci/l)), predominantly from Cs 137; about 80,000 gal/yr (300,000 l/yr) is produced. The mix used throughout consists of Portland cement (38.5 wt%), fly ash (38.5 wt%), Attapulgite drilling clay (15.4 wt%), clay (7.7 wt%), and a retarder (0.05 wt%). The cost of this mix is about 10 cents per gallon (2.6 cents/l) of waste and the leach resistance is comparable to that of a good borosilicate glass. The normal flow rate of waste solution is about 180 gal/min (680 l/min); normal mix ratio is about 7.5 lb of solids per gallon of waste (0.9 kg/l); and normal grout injection rate is about 250 gal/min (950 l/min). An injection takes about 8 hours to complete. A few weeks after injection, the well is opened and any free water that has separated from the grout is collected and fed to the Laboratory waste collection system. The bleed-back water usually contains less than 0.1% of the injected radionuclides. General experience with the shale fracturing facility in 18 operating injections has been quite good. Only one injection had to be terminated (the result of an attempt to use blended soils that had been stored for several months and showed poor flowability), and only one delayed (due to failure of a packing seal on the injection pump). A new facility is being planned to allow handling of waste solutions and sludges with higher specific activities [up to 20 Ci/gal (5.3 Ci/l)]. (LKH)

Alkalinity; Specific Activity; Waste Volume

HYDRAULIC FRACTURING; SHALES; SEDIMENTARY ROCKS; WASTES, INTERMEDIATE-LEVEL; CEMENTS; GROUTING; WELLS, INJECTION; METHODS; MONITORING; WASTE VOLUME; SPECIFIC GRAVITY; ALKALINITY; CESIUM 137; ATTAPULGITE; CLAYS; COSTS; FLOW RATE; CHEMICAL COMPOSITION; SLUDGES; WASTE DISPOSAL; STORAGE, GEOLOGIC; WASTE MANAGEMENT; FIELD STUDIES

<214>
Weeren, H.O., J.G. Moore, and E.W. McDaniel, Oak Ridge National Laboratory, Oak Ridge, TN.

Waste Disposal by Shale Fracturing at Oak Ridge National Laboratory.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 257-260), 563 pp. (1979)

Shale fracturing is a process currently being used at the Oak Ridge National Laboratory for the permanent disposal of locally generated intermediate-level radioactive waste solutions. In this process, the waste is mixed with a solids blend of Portland cement (the overall binder), fly ash (to fix strontium) drilling clay (to retain excess water), pottery clay (to fix cesium), and a retarder (to delay setting time of the grout). The resulting grout is then injected down a well into an impermeable shale formation at a depth of 200 to 300 m (700 to 1000 ft), underlain by a dense sandstone and well below groundwater levels, where it hardens into a thin horizontal grout sheet several hundred feet across. One 8 hr injection disposes of one year's accumulation of waste solution that has been stored in tanks at the site prior to injection. A network of observation wells surrounds the facility. Injection pressure and radiation exposure to the operating crew are monitored during each injection. After several injections, the cumulative surface uplift around the well is determined and the continued impermeability of the shale is verified. Leach rates were measured in these studies, and were found to be diffusion controlled. For cesium, observed values of the diffusion coefficient ranged from 10(E-12) to 10(E-10) sq cm/sec. Strontium had a typical value of 5 X 10(E-12), curium

<218>

DISPOSAL SITE

<218> CONT.
10(E-16), and plutonium 10(E-17) sq cm/sec. The essential feature of the shale fracturing process is the fixation of radionuclides in a geological formation known to be isolated from contact with the surface environment. Leach rates of significant radionuclides from the set grout are low, but any radionuclides that might be leached would be retained in the disposal zone by the high ion exchange capacity of the shale. (Auth) (DCN)

Diffusion Coefficient; Injection Pressure

Cs; Sr; Pu; Cm

WASTES, RADIOACTIVE; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTE DISPOSAL; WASTE STORAGE; STORAGE, FACILITIES; STORAGE, GEOLOGIC; DISPOSAL SITE; SANDSTONES; SHALES; SEDIMENTARY ROCKS; GEOLOGIC FORMATIONS; GROUTING; FRACTURES; INJECTION; WELLS, INJECTION; LEACHING; DIFFUSION COEFFICIENT; MONITORING; FIXATION; METHODS; FIELD STUDIES; CESIUM; STRONTIUM; PLUTONIUM; CURIUM

<215>
Ahlstrom, S.W., H.F. Foote, R.C. Arnett, C.R. Cole, and R.J. Serne, Battelle-Pacific Northwest Laboratories, Richland, WA; Atlantic Richfield Hanford Company, Richland, WA.

Multicomponent Mass Transport Model: Theory and Numerical Implementation (Discrete-Parcel-Random-Walk Version).

BNWL-2127; 104 pp.(1977, May)

The Multicomponent Mass Transfer (MMT) Model is a generic computer code, currently in its third generation, that was developed to predict the movement of radiocontaminants in the saturated and unsaturated sediments of the Hanford Site. This model was designed to use the water movement patterns produced by the unsaturated and saturated flow models coupled with dispersion and soil-waste reaction submodels to predict contaminant transport. This report documents the theoretical foundation and the numerical solution procedure of the current (third) generation of the MMT Model. The present model simulates mass transport processes using an analog referred to as the Discrete-Parcel-Random-Walk (DPRW) algorithm. The basic concepts of this solution technique are described and the advantages and disadvantages of the DPRW scheme are discussed in relation to more conventional numerical techniques such as the finite-difference and finite-element methods. Verification of the numerical algorithm is demonstrated by comparing model results with known closed-form solutions. A brief error and sensitivity analysis of the algorithm with respect to numerical parameters is also presented. A simulation of the tritium plume beneath the Hanford Site is included to illustrate the use of the model in a typical application. (Auth)

COMPUTER PROGRAMS; CONTAMINATION; DIFFUSION; DISPERSIVITY; FLUID MECHANICS; GROUND WATER; HYDRODYNAMICS; MODELS, MATHEMATICAL; NUMERICAL METHODS; RADIONUCLIDE MIGRATION; SATURATED ZONE; UNSATURATED ZONE; EQUATIONS; THEORETICAL STUDIES

<216>
Alphen, E.L., Battelle-Pacific Northwest Laboratories, Division of Production and Materials Management and Hanford Plant Assistance Programs, Richland, WA.

Pacific Northwest Laboratory Monthly Activities Report January 1978.

BNWL-1818; 29 pp.(1978, March)

Work carried out in January 1978 as assistance for Atlantic Richfield Hanford Company included testing of transducers experimentally exposed to gamma radiation; development of a CuCl capsule leak testing system; heat transfer studies on solidification of ITS recycle solutions by phosphoric acid addition; evaluation of corrosion rate probes for in-tank corrosion rate measurements; evaluation of cathodic protection of tank liners; review of a fault-tree for examining potential modes of failure in the waste tank farm system; generation of a tank leak model; review of the present tank leak detection system; study of ultrasonic logging and radio frequency detection systems for liquid level measurements in storage tanks; and evaluation of soil resistivity as a leak detector. Other studies were conducted in the areas of Pu scrap processing development, Pu-soil relationships at the 216-Z-9 trench, tank farm support, surficial contamination and airborne release studies, groundwater management, and biological transport. Technical assistance to the Hanford Plant in January consisted of continuation of the Routine Environmental Surveillance Program, (all measurements were within expected ranges), programming of phase three of the in-vivo Radiation Protection computer program, operation and maintenance training for new bench-monitors delivered to a Hanford contractor, and evaluation of a neutron counting system and alpha floor monitor. Sections abstracted separately for this data base are Plutonium Trench Soil Chemistry and Mineralogy, and Generation of Improved Transmissivity Distribution. (LKM)

Leaching Rate

Pu

MAPS; AQUIFERS; GEOLOGY; HYDROLOGY; TRANSMISSIVITY; GROUND WATER; MODELS, MATHEMATICAL; COMPUTER PROGRAMS; SOILS; PLUTONIUM; SORPTION; DESORPTION; LEACHING; POLYMERS; AUTORADIOGRAPHY; TRENCHES; BURIAL; WASTE DISPOSAL; SALTS; DISPOSAL SITE

<217>
Ames, L.L., and D. Rai, Battelle-Pacific Northwest Laboratories, Richland, WA.

Radionuclide Interaction with Soil and Rock Media. Volume 1: Processes Influencing Radionuclide Mobility and Retention; Element Chemistry and Geochemistry; Conclusions and Evaluation.

EPA 520/6-78-037; 307 pp.(1978, August)

The interactions of 19 radionuclides with soil and rock media in the literature to 1977 are reviewed. The radionuclides are those of the elements As, Sb, Ce, Cs, Co, Cm, Pu, I, Pb, Pa, Ra, Ru, Sr, Tc, Th, H3, U, and Zr. The review includes a brief summary of natural soil and rock distributions, chemistry, solid and solution equilibria and laboratory field adsorption and migration results. It is concluded that most of the values for radionuclide-geologic media distribution coefficients in the literature are inadequate due to failure to include a comprehensive description of system conditions. Evaluation of the literature reveals that soils and sediments mainly show cation exchange capacities; thus most cations migrate through soil or rock at speeds slower than groundwater. Relative to each other, trivalent cations generally move the slowest, followed by the divalent and monovalent cations. Tritium readily substitutes for hydrogen in water and therefore migrates at the same velocity as water. Complicating factors include a higher selectivity of a soil or rock component for a given cation or a more successful hydrogen ion competition with one cation versus another. The relative mobility of Sr over Cs at low pH is an example of the latter. Simple anions tend to migrate with little reaction because of a pH less than 8 is usually required to activate a significant soil anion exchange capacity. However, both I and Tc, which exist primarily as anions, undergo reactions with organic ligands that can greatly slow their migration. The migration and retention of

<217>

ENVIRONMENTAL TRANSPORT

<217> CONT.

inorganic complex species would also be dependent upon the charge and size of the species, with positively charged species moving slower than negatively charged species. The highly charged metal ions tend to polymerize or form colloidal-size, charged precipitates. The trivalent metal ions (Am, Cm, Sb, Bm) and quadrivalent or higher charged metal ions (Pu, Th, U, Zr) are known to form polymers. Initially, at lower pH, the polymers are positively charged and at pH 8 become increasingly negatively charged, resulting in good adsorption and very slow movement at acidic to neutral pH, and greatly lessened adsorption and rapid migration from pH 8 upward. The behavior of organic-complexed species of elements is difficult to predict because of a lack of knowledge regarding the exact nature of the organic ligands, a wide variation in amounts and types of organic ligands, and the size and solubility of these organics. The metal size and charge of the species involved and the ability of the final mineral ions that take part in replacement reactions depend upon the size and charge of the species involved and the ability of the final mineral product to accept that species into its growing structure. These metal ions are buried in the replacement product crystal structure and are effectively retained unless a changing chemical environment causes instability and dissolution of the crystal. Areas for future work suggested by this review include determination of adsorption mechanisms, and kinetics, influence of redox potential on Sb, I, Np, Pu, Tc, and U migration potential, influence of organic ligands on radionuclide migration potential, thermodynamic data determinations of solid and solution species, generation of distribution coefficient data than can be used in a comparative mode, and maintenance of the data in a bank with general access available. (Auth) (LKH)

WASTE-ROCK INTERACTIONS; RADIONUCLIDE MIGRATION; REVIEWS; DISTRIBUTION COEFFICIENT; EVALUATION; CATION EXCHANGE CAPACITY; SOILS; ROCKS; RADIONUCLIDES; GEOCHEMISTRY; EQUILIBRIUM CONSTANT; ADSORPTION; SEDIMENTS; CATIONS; GROUND WATER; VALENCE; pH; ANIONS; ION EXCHANGE CAPACITY; ORGANIC COMPOUNDS; COMPLEXES; COLLOIDS; POLYMERS; RECOMMENDATIONS

<218>

Apps, J.A., N.G.W. Cook, and P.A. Witherspoon, Lawrence Berkeley Laboratory, Berkeley, CA.

An Appraisal of Underground Radioactive Waste Disposal in Argillaceous and Crystalline Rocks: Some Geochemical, Geomechanical, and Hydrogeological Questions.

LBL-7047; Geotechnical Assessment and Instrumentation Needs in Crystalline and Argillaceous Rocks for Radioactive Waste Storage, Proceedings of a Symposium, Berkeley, CA, July 16-20, 1978, (50 pp.).(1978, June)

An appraisal of radioactive waste disposal in argillaceous and crystalline rocks is attempted, emphasizing mechanical, hydrologic, and geochemical considerations. The geochemical stability of a suitable repository site depends in general on the absence of major geological perturbations, such as earthquake faults and volcanism, and, specifically, on a state of stress conducive to stability. A conservative interpretation of existing evidence suggests that the value of the maximum principal stress should not exceed 25 MPa and that the value of the minimum principal stress should be no less than two-thirds of this. A lithostatic state of stress at a depth of a kilometer would constitute a near ideal condition. The mass of a radionuclide decreases by about nine orders of magnitude over a period of thirty half lives. If the product of the concentration of a radionuclide in the groundwater and the total amount of groundwater contaminated with this radionuclide is small over such a period, the duration of leaching is controlled by radioactive decay. This is an important criterion for the design and selection of waste forms. For wide ranges of radionuclide concentrations leaving a repository, from 10(E-3) to 10(E-12) (g mol/kg), the distances over which the radionuclides may move during a million years can be insignificant except in the case of suspended particulates with concentrations of 10(E-3) or more. Given joints and fractures of sufficently low hydraulic conductivity, small hydraulic gradients and long flow paths, the time taken for water from a repository to reach the biosphere may be of the order of a million years. This is regarded as a sufficient period of time for isolation. The results of this appraisal suggest that argillaceous and crystalline rocks may provide suitable sites for nuclear waste repositories. The most important factor affecting their suitability appears to be the flux of groundwater through the repository. Although intense fracturing would be advantageous in retarding migration of hazardous materials from a repository in these rocks by sorption, the hydraulic conductivities of these fractures would have to be very small and the hydraulic gradient low. (Auth) (LKH)

REVIEWS; GEOCHEMISTRY; HYDROLOGY; RADIONUCLIDE MIGRATION; ROCK MECHANICS; REPOSITORY; STRESS; GROUND WATER; CONCENTRATION; PARTICLES; ADSORPTION; JOINTS; FRACTURES; HYDRAULIC CONDUCTIVITY; HYDRAULIC GRADIENT; SEDIMENTARY ROCKS; IGNEOUS ROCKS; THEORETICAL STUDIES; WASTE MANAGEMENT; WASTE DISPOSAL; LEACHING

<219>

Apps, J.A., J. Lucas, A.K. Mathur, and L. Tsao, Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA.

Theoretical and Experimental Evaluation of Waste Transport in Selected Rocks: 1977 Annual Report of Lawrence Berkeley Laboratory Contract No. 85901AK. Waste Isolation Safety Assessment Program - Collection and Generation of Transport Data.

LBL-7022; 139 pp.(1977, September)

This report summarizes progress and results obtained during FY 1977 as part of Subtask 4 of Task 5 of the Waste Isolation Safety Assessment Program (WISAP) managed by Battelle Pacific Northwest Laboratories. The goal of this project is to establish a basis on which radionuclide distribution coefficients can be reliably predicted for geologic environments of the type anticipated for terminal radioactive waste storage facilities. The radionuclides chosen for this study include Th, U, Np, Pu, Am, Cm, I, and Tc; during FY 1977, emphasis has been placed on Np, Pu, Am, Cm. Host rock types being considered include acid igneous rocks

<219> CONT.
(granite, rhyolite), basic igneous rocks (gabbro, dolerite, and basalt), sedimentary rocks (argillites, sandstone, and limestone), and metamorphic rocks (if deemed appropriate). Although the study is generic, greater emphasis is being placed on rocks from the Nevada Test Site and the Hanford Reservation in Washington. During FY 1977, the following subtasks were performed: 1) thermodynamic data were tabulated for those aqueous complexes and solid phases of Np, Pu, Am, and Cm likely to form in the natural environment; 2) Eh-pH diagrams were computed and drafted for Np, Pu, Am, and Cm at 25 C and 1 atmosphere pressure; 3) the literature on distribution coefficients of Np, Pu, Am, and Cm was reviewed; 5) preliminary considerations were determined for an experimental method of measuring radionuclide transport in water-saturated rocks; 5) the transport mechanisms of radionuclides in water--saturated rocks were reviewed; and 6) a computer simulation was attempted of mass transfer involving actinides in water-saturated rocks. Subtasks 1, 2, 3, and 4 are complete. Progress in Subtask 5 is represented by an initial theoretical survey to define the conditions needed to characterize the transport of radionuclides in rocks. This task will be refined and continued in FY 1978. Subtask 6 has begun but is not complete; its progress will be reported more fully in 1978. (auth) (LKN)

DISTRIBUTION COEFFICIENT; STORAGE, GEOLOGIC; RADIONUCLIDES; IGNEOUS ROCKS; SEDIMENTARY ROCKS; METAMORPHIC ROCKS; GRANITES; RHYOLITES; GABBROS; DOLERITES; BASALTS; SHALES; SANDSTONES; LIMESTONES; THERMAL PROPERTIES; Eh; pH; REVIEWS; RADIONUCLIDE MIGRATION; SATURATED ZONE; GROUND WATER; MODELS; MODELS, MATHEMATICAL; COMPUTER PROGRAMS; ACTINIDES; THEORETICAL STUDIES

<220>
Apt, K.E., R.A. Bryant, and E.G. Brookins, Los Alamos Scientific Laboratory, Los Alamos, NM; University of New Mexico, Albuquerque, NM.

Oklo - Geologic Isolation Implications.

CONF-761020; LA-UR-76-2096; Waste Management '76, R.G. Post (Ed.), Proceedings of a Symposium, Tucson, AZ, October 3-6, 1976, (pp. 200-205), 370 pp. (1976)

The Oklo "natural reactor" is discussed as an example of successful geologic isolation of radioactive wastes. The deposit is located in Gabon, Africa, in the middle Proterozoic Franceville Basin. U mineralization is within a stratiform ore-body, with 6 tabular "reactor zones" recognized, each about 1 m thick and several hundred sq m in area. Maximum observed U concentrations are around 60%, corresponding to an original 80% - very near the 85% stoichometric limit for U_3O_8. The orebody was probably water-saturated initially, and then began fission in over-moderated state. The power level of the reactor tended to optimize neutron moderation by driving off water via thermal expansion or possibly bubble formation. Temperatures and pressures did not exceed the critical point of water. Criticality was sustained for approximately 0.6 my at about 1.8 + or - 0.1 by ago. About 6 tons of U 235 burned, along with fast neutron-induced fission of small amounts of U 238 and thermal neutron-induced fission of Pu 239. Average power level for the duration of the reaction was 25 kW/yr. Of the approximately 6 tons of fission products formed, the majority remain in place. The lanthanides were almost completely immobile; order of stability in the reactor zone was Nd greater than Sm greater than Gd. Ru depletion is less than 30%; Ru and Tc appear to have been redistributed somewhat within the reactor zone, as do Zr and Pb. Mo is about 90% depleted. The alkali metals and alkaline earths tended to migrate, only a few percent remaining in the reactor zone. Th, U, Np, and Pu were almost completely isolated in the reactor system, with only some redistribution of U. Two tons of Pu 239 produced via U 238 neutron/gamma decay did not disperse; most underwent alpha decay. This stability of Pu is best explained by the reducing environment of the ore deposit. Successful isolation of this deposit resulted from two factors: the relative tectonic stability of the Franceville Basin during the period when the wastes were most hazardous, and the effectiveness of the geochemical environment as a barrier to migration. (LKN)

Table 1 gives the migration of reactor products in the zone 2 reactor at the Oklo mine.

GEOLOGIC DEPOSITS; RADIONUCLIDE MIGRATION; URANIUM; REACTORS; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; WASTE MANAGEMENT; WASTE DISPOSAL; GEOLOGIC DEPOSITS; PLUTONIUM; THORIUM; NEPTUNIUM; FISSION PRODUCTS; FIXATION; IMMOBILIZATION; LABORATORY STUDIES; Eh; pH

<221>
Arnett, R.C., D.J. Brown, and R.G. Baca, Atlantic Richfield Hanford Company, Richland, WA.

Hanford Groundwater Transport Estimates of Hypothetical Radioactive Waste Incidents.

ARH-LD-162; 30 pp. (1977, June)

A study was made to evaluate the environmental consequences along the groundwater flow path which may result from hypothetical leaks from the Hanford high-level radioactive waste tanks. A series of waste tank accidental discharges were postulated, including the sudden release of 800,000 gallons of high-level waste, the release of 680,000 gallons of high-level waste from a tank over a period of 3,400 days, and the release of 100,000 gallons from a series of 4 adjacent tanks over a period of 125-150 days. All hypothetical leaks were for tanks located in the 200 East area. The ground water flow and contaminant transport models used for predicting the radionuclide movement in the uppermost or confined ground water flow system were the Variable Thickness Transient Groundwater Flow Model and the Multicomponent Mass Transport Discrete Parcel Random Walk Model. The analyses indicate that if the entire volume of a tank were to be released to the ground, it is unlikely under normal circumstances that any of the liquid or associated contaminants would reach the water table. If it is assumed that substantial fractions of the material reached the ground waters, the projected spatially averaged concentrations of Tc 99, H 3, and Ru 106 in the ground water resulting from the hypothetical tank leaks upon discharge into the Columbia River would be far below the respective ERDA Manual Chapter 0524 Concentration Guides. Upon entering the Columbia River, additional large dilutions of

<221>

ENVIRONMENTAL TRANSPORT

<221> CONT.
the slightly contaminated water would further reduce any biological risk. (JRT)

Tc 99; H 3; Ru 106

ACCIDENTS; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; TANKS; HYDROLOGY; GROUND WATER; CONTAMINATION; TRITIUM; RUBIDIUM; TECHNETIUM; WATER TABLE; MODELS, MATHEMATICAL; SAFETY; HAZARD ANALYSIS; RADIONUCLIDE MIGRATION; THEORETICAL STUDIES

<222>
Arnett, R.C., R.E. Gephart, K.A. Deju, C.R. Cole, and S.M. Ahlstrom, Atlantic Richfield Hanford Company, Richland, WA.

Hanford Groundwater Scenario Studies.

ARH-SA-292; 83 pp. (1977, May)

This report documents the results of two Hanford groundwater scenario studies. The first study documents the hydrologic impact of increased groundwater recharge resulting from agricultural development in the Cold Creek Valley located west of the Hanford Reservation. The second study involves recovering liquid radioactive waste which has leaked into the groundwater flow system from a hypothetical buried tank containing high-level radioactive waste. The predictive and control capacity of the onsite Hanford modelling technology is used to evaluate both scenarios. Modelling the Cold Creek Valley Irrigation scenario consisted of two computer runs of the Variable Thickness Transient Model; one 5-year transient run, and one steady state run. The Multicomponent Mass Transport Discrete Parcel Random Walk Model was used to simulate the 5-year tritium movement using the transient flow field and the steady state flow field. The Hanford Pathline Calculational Program Model was used to compare the basic streamline and travel times of water moving from the 200 West Area to the Columbia River. The results show that at the model boundary within the Cold Creek Valley the water table would increase about 30 ft and range down to 1 ft or less on a north-south line passing through the east end of Gable Butte and on an east-west line passing through the west end of Gable Butte. The maximum predicted rise in the water table beneath the 200 West Area was 6-11 ft; the rise in water table beneath 200 East Areas was less than 1 ft. The results of the Cold Creek Valley Irrigation Study indicate that heavy agricultural irrigation in the valley is unlikely to cause significant changes either in the water table underlying the high-level waste areas or in the movement of radionuclides already in the groundwater. Although some limitations of the present Hanford groundwater modelling capability were identified for this type of comparison, such limitations are of secondary importance and are unlikely to affect the overall conclusion. Simulation of the hypothetical tank leak used the same models. The waste was assumed to consist of 250,000 gal of liquid with the same properties as water and containing over 87,000 Ci of Ru 106. The waste was assumed to have spread over a 30-day period before recovery operations began, diluting it to about 7,200,000 gal. The most effective recovery configuration determined from the simulation consisted of 3 wells 30-35 ft apart in a triangular pattern with one well at the leak site and the other two downgradient. Simulation of pumping at 150 gpm and 325 gpm demonstrated that cleanup was accelerated with the higher rate, but that the recovered waste solution was of higher volume. Total groundwater contamination was reduced to 5770 Ci, a reduction factor of approximately 8. The hypothetical tank study showed that an active response (in this case waste recovery) can be modelled and is a possible alternative to passive monitoring of radionuclide movement in the unlikely event that high-level waste is introduced into the groundwater. However, the usefulness of the existing modelling capability for such simulations is limited. Research is now underway to correct these weaknesses. (Auth) (LKH)

MODELS; MODELS, MATHEMATICAL; COMPUTER PROGRAMS; COMPUTERS; HYDROLOGY; RECHARGE; AGRICULTURE; WASTES, RADIOACTIVE; WASTES, LIQUID; AQUIFERS; TANKS; WASTES, HIGH-LEVEL; PREDICTIONS; TRITIUM; RADIONUCLIDE MIGRATION; WATER TABLE; IRRIGATION; WELLS; WITHDRAWAL; THEORETICAL STUDIES; DISPOSAL SITE; LEAKS

<223>
Balestri, R.J., and B.W. Terres, The BDM Corporation, Albuquerque, NM.

The Utilization of the HOL Data Structure in Consequence Analysis for WIPP Repository.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 29-December 1, 1978, (pp. 72-73), 77 pp. (1978)

To facilitate the safety analysis of the WIPP a computer code was developed around a high order language (HOL), an attribute linked data structure, and interchangable nuclides. Of the three safety analysis methods, risk analysis, cost benefit analysis, and consequence analysis, only the latter is meaningful. The SLED program performs consequence analysis by comparing the integrated dose in a population or individual to the NRC guidelines for industrial nuclear workers. Additionally the response by organ is determined. The data structure contains site, inventory, response, life, biological and geological data. Parametric studies can be performed on any element of any data structure. (NDV)

This paper was presented at the conference but was not included in the Proceedings.

WIPP; HAZARD ANALYSIS; SAFETY; COMPUTER PROGRAMS; DOSE COMMITMENTS; STATISTICS

<224>
Barnhart, B.J., Los Alamos Scientific Laboratory, Los Alamos, NM.

Potential Microbial Impact on Transuranic Wastes.

LA-7582-PR; Studies of Transuranic Waste Storage under Conditions Expected in the Waste Isolation Pilot Project (WIPP), July 1-September 30, 1978, (pp. 13-15) 16 pp. (1979, January)

Progress is reported for the first quarter of a project to assess potential microbial interaction with plutonium and other

<224> CONT.
transuranics in low-level radioactive waste as it is expected to occur in the Waste Isolation Pilot Project (WIPP). The initial effort is to enumerate and categorize morphologically and biochemically the naturally occuring microorganisms found in soils and in representative samples of TRU wastes isolated in steel and wooden containers in a 0-10 yr old and 25-30 yr old shallow burial site at LASL. This phase of the project will also include detection of chelated TRU radioisotopes and possible microbial production of such chelates. The second phase will be a longer term assessment of the influence of experimentally inoculated microorganisms on the chemical and physical state of TRU wastes, including Pu isotopes, and the biochemical and cytotoxic influence of these radioisotopes on the microbial population. In the two months this project has been funded, an extensive body of pertinent literature has been compiled, equipment and laboratory expendables have been ordered and received, enumeration, isolation, and identification of bacteria and fungi from soil obtained from the Area G drum-burial site has been initiated, and abiotic alkylation of a heavy metal (mercury) has been successfully performed, and the methylated product identified in preparation for subsequent abiotic and biological alkylation reaction of europium and plutonium. Work on the second phase of the project, which will be dependent on results of the first phase, will be ongoing throughout FY 1979. (Auth) (LKM)

WIPP; MICROORGANISMS; BACTERIA; PLUTONIUM; EUROPIUM; TRANSURANICS; WASTES, LOW-LEVEL; WASTES, TRANSURANIC; CHELATES; RADIATION EFFECTS; CHEMICAL PROPERTIES; SOILS; WASTE STORAGE; STORAGE, GEOLOGIC; LABORATORY STUDIES

<225>
Bartlett, J.W., Battelle-Pacific Northwest Laboratories, Richland, WA.

Development of Advanced Waste Management Plans.

BNWL-1978; Nuclear Waste Management and Transportation Quarterly Progress Report, October through December 1975 (pp. 1-8), 15 pp. (1976, March)

The Advanced Waste Management Program was initiated in 1972 as an overview planning, concept development, and systems analysis function of the Division of Waste Management and Transportation. The program previously consisted of three tasks: development of R&D plans, development of evaluation methods, and special studies. Plans for future work involve two major tasks: A) final storage/disposal of nuclear waste and B) assessment of alternative fuel-cycle and waste management systems. Task A will emphasize (through FY 1977) development of a plan for identification of several acceptable final geologic storage sites, assessment of criteria and alternative technologies to implement the plan, and assessment of the implications and constraints of retrievability. Also, methods will be developed and applied for evaluation of barriers in alternative disposal systems (systems aspect to begin FY 1976), including projection of radionuclide migration. Conceptual radionuclide transport models for geologic media have been completed by PNL, but have not been tested due to lack of data. A detailed work plan for activities in this area is being developed, with a workshop on data requirements scheduled for February 1976 and plans for a companion workshop on models. Task B objectives are continuing compilation and description of alternative means of handling the post-fission part of the LWR fuel cycle, and provision of analytical methods and results for assessment of those systems. This task is currently inactive as work continues on the Technical Alternatives Document (TAD). Work on TAD has produced the comprehensive generic concept of alternative systems, which is the basis for development of the analytical methods. Application of TAD to this task is expected to begin in CY 1976. (LKM)

STORAGE, GEOLOGIC; RADIONUCLIDE MIGRATION; MODELS; MODELS, MATHEMATICAL; GEOLOGIC STRATA; WASTE MANAGEMENT; WASTE DISPOSAL; WASTE STORAGE; FUEL CYCLES; METHODS; SITE SELECTION; WASTES, RADIOACTIVE; RETRIEVABILITY

<226>
Bartlett, J.W., Battelle-Pacific Northwest Laboratories, Richland, WA.

Nuclear Waste Management and Transportation Quarterly Progress Report, October Through December 1975.

BNWL-1978; 15 pp. (1976, March)

Work in this program has been divided into two tasks: 1) Development of Advanced Waste Management Plans, and 2) Development of Evaluation Methodology. Plans for work Task 1 involve development and assessment of plans for final storage/disposal of nuclear waste and assessment of alternative fuel cycle and waste management systems. Under Task 2, a model to describe diffusion-controlled migration of radionuclides from a repository in geologic formations was developed. In the new model, convective transport is ignored and the molecular diffusivities of the various chain members are assumed to be different; the boundary conditions for impulse and band releases are the same as in the previous model. A computational scheme is now being prepared. Models of radionuclide migration in faulted media were also developed and found to differ from models for homogeneous media only in the numerical values for dispersion coefficient and in the relationship for calculating the scheme sorption equilibrium constant. A report on surveys of public attitudes toward waste management alternatives was completed and issued for internal review. Four articles have been separately entered into the data base. (LKM)

RADIONUCLIDE MIGRATION; WASTE MANAGEMENT; WASTE STORAGE; WASTE DISPOSAL; FUEL CYCLES; STORAGE, GEOLOGIC; FAULTS; DISPERSION COEFFICIENT; SORPTION; MODELS; MODELS, MATHEMATICAL; WASTES, RADIOACTIVE

<227>
Bochkarev, V.N., Z.G. Antropova, and T.I. Belova, Institute of Biophysics, U.S.S.R.

Migration of Strontium 90 and Cesium 137 in Soils of Different Texture.

<227>

ENVIRONMENTAL TRANSPORT

<227> CONT.
Soviet Soil Science (9):936-938.(1965, April)

The migration of Sr 90 and Ce 144 through the profile of medium and coarse clay loam podzolized chernozems was investigated. Experiments were performed on two plots on fields with flat smooth relief, uncultivated for 6-7 years. Each plot area was 5 x 5 sq m. A mixture of Sr 90 and Ce 144 in the form of a nitrate solution with minimum content of stable isotopic carriers was applied to the surface of these plots; density of contamination was 3 mc/sq m for Ce 144 and 0.3 mc/sq m for Sr 90. Soil samples were taken after 2 and 3 years. Five profiles were dug to a depth of 40 cm on each plot to obtain soil samples and two profiles were dug to 150 cm to describe soil characteristics. Since soil activity below 10 cm was caused wholly by natural K 40, samples below this level were not taken for subsequent investigations. Radiochemical determination of the isotopes was carried out by fusing oven-dry soil with potassium and sodium carbonates in a muffle oven at 750-800 C for 2.5-3 hrs. To eliminate silicic acid, the soil was first treated several times with HF. The fluoride ions in solution were then removed by treatment several times with HF and then with a saturated solution of oxalic acid. The oxalates thus created are easily destroyed by heating in a muffle oven at 450-500 C. The fusion was dissolved in HCl; Sr 90 was determined in one part of the solution, Ce 144 in the other. Two years after the radioisotopes were added to coarse clay loam chernozem, their activity at a depth of 4-6 cm was 3 times greater than in medium clay loam chernozem at the same depth, and a year later at 14-16 cm it was 20 times greater. If the total activity distributed through the profile is taken as 100%, 15% is concentrated in the coarse clay loam chernozem below 5 cm as compared with only about 4% in the medium clay loam. The distributions of Sr 90 and Ce 144 obtained for the medium clay loam samples were as follows: 0-1 cm depth = 2.34 Ci/kg Sr 90, 5.43 Ci/kg Ce 144; 4-6 cm = 3.24 Ci/kg Sr 90, 5.3 Ci/kg Ce 144; 9-11 cm = 9.9 Ci/kg Sr 90, 2.2 Ci/kg Ce 144; 14-16 cm = 3.5 Ci/kg Sr 90, 2.9 Ci/kg Ce 144. Distributions in coarse clay loam were: 0-1 cm depth = 1.94 Ci/kg Sr 90, 3.76 Ci/kg Ce 144; 4-6 cm = 9.7 Ci/kg Sr 90, 1.24 Ci/kg Ce 144; 9-11 cm = 1.0 Ci/kg Sr 90, 7.8 Ci/kg Ce 144; 14-16 cm = 4.73 Ci/kg Sr 90, 1.43 Ci/kg Ce 144. The results show that Sr 90 is relatively more mobile than Ce 144. At a depth of 14-16 cm in coarse clay loam chernozem, Sr 90 activity constituted 60-70% of the total activity as compared with 33.3% for Ce 144. (LKH)

Total Ion Concentration

Sr 90; Ce 144

SOILS; STRONTIUM 90; CERIUM 144; LOAM; CLAYS; NITRATES; SAMPLES; METHODS; LABORATORY STUDIES; FIELD STUDIES; RADIONUCLIDE MIGRATION; SOIL TRANSPORT

<228>
Bondietti, E.A., and C.W. Francis, Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Chemistry of Technetium 99 and Neptunium 237 in Contact with Unweathered Igneous Rocks.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 417-418), 563 pp.(1979)

Two radioelements with long half-lives and relatively mobile oxidation states are technetium and neptunium. The oxidation state species commonly used experimentally are TcO_4^- and NpO_2^+ since these are the stable states in the presence of air. Both TcO_4^- and NpO_2^+, however, are reduced to less water-soluble species when contacted with unweathered Sentinel Gap basalt, Climax Stock granite, and Westerly granite under conditions where O2 is carefully excluded. The instability of both NpO_2^+ and TcO_4^- was predicted by comparing thermodynamic calculations with actual redox (Eh) measurements of the rock-water systems. The actual changes in oxidation states were evaluated during the experiments by two methods. Technetium 99 less than or equal to 10(E-6) M rapidly disappeared from solution in the presence of all three igneous rocks when equilibrations were conducted in the absence of air. If the rocks were artificially weathered, no Tc was lost from solution. Technetium could not be recovered from the rocks by leaching with 2 M NaCl but was partially recovered if an oxidant (BrO_3^-) was present. Neptunium (V) itself is adsorbed by the rocks. However, the presence of Np (IV) was unambiguously determined both in the solution-phase and in association with the rocks. These results indicate that highly insoluble TcO2 and NpO2 are the likely forms influencing solution concentrations of these two elements in anoxic ground waters. While it is possible to predict this from thermodynamic considerations alone, our results represent experimental verification and thus mitigate the paucity of (geochemical) experimental data on these two elements. (Auth)

Abstract is complete text.

TECHNETIUM; NEPTUNIUM; GRANITES; BASALTS; WASTE-ROCK INTERACTIONS; ADSORPTION; Eh; LABORATORY STUDIES

<229>
Bouvard, P., A. Grauby, and S.D. Blalock, Jr. (Translator), Commission for Atomic Energy, Radiation Control Service, France; Commission for Atomic Energy, Radioactive Products Control Service, France.

A Study of the Migration of Radioactive Elements in Soils "In-Place".

ORNL/tr-4681; Retention and Migration of Radioactive Ions in Soils, Proceedings of an International Symposium, Saclay, France, October 16-18, 1962, (pp. 123-129).(1962, October)

This paper presents a new method for study of radionuclide migration in samples of undisturbed soils. A column of soil 20 x 30 cm is dug out, sealed, and returned to the lab, where the contaminating solution is added. Volume and radioactivity of the eluate are measured after passing through the column, and a vertical section of the soil is autoradiographed. These data are then related to hydrodynamic characteristics, humidity, and mineralogy. For the doses used, radioactivity remains mainly in the first

<229> CONT.
15mm of soil. Dry soils tend to retain more of the radionuclide than wet ones. Carbonate soils and iron-rich soils tend to have a higher retention than non-carbonate, non-iron soils. 95-100% fixation of 5-6 uCi doses of Sr per mass were obtained in an iron-rich hydromorphic soil. Leaching tests on various soil types showed that in podzol, spread of the leading edge of contamination is slow but proportional to intensity of leaching; in dark and gley soils, the leading edge of contamination advances to about 20 cm, regardless of leaching intensity. (LKM)

Moisture Content; Leaching Rate

Sr 90; Y 90

SOILS; RADIONUCLIDE MIGRATION; STRONTIUM; YTTERBIUM; AUTORADIOGRAPHY; RADIOACTIVITY; LEACHING; LABORATORY STUDIES

<230>
Bowen, V.T., Woods Hole Oceanographic Institution, Woods Hole, MA.

Oceanic Distribution of Radionuclides from Nuclear Weapons Testing.

Transactions of the American Nuclear Society 26:297; CONF-770611; American Nuclear Society Annual Meeting, New York, NY, June 12-16, 1977, (p. 297). (1977)

Movement in the oceans of atmospheric fallout from nuclear weapons testing provide information applicable to studies of seabed disposal of radioactive wastes. Some of the long-lived nuclides move in the oceans largely as solutes, others largely after association with particles, but a variety of intermediate behaviors can be discerned. Sr 90 and Cs 137, traveling predominantly in solution, show strong effects, even after decades, of the latitudinally dependent pattern of their original delivery to the earth's surface. Pu, Am 241, and Fe 55 move largely with particulate matter. Regional conditions have produced variation in retention of some of these nuclides in water columns. Their rates of sinking, estimated from sediment inventories under water columns of various depths, permits the definition of several classes of nuclide-bearing particles with different sinking rates and with different affinities for these three tracers. Fixation rates of these particles are likely influenced by local differences in bioturbation. Once incorporated in marine sediments, Pu, Am, Fe, and Cs are moved downward to substantial depths by complicated processes of biological mixing and water circulation. In shallow-water sediments, however, differences in the elements' chemical behavior result in the return of each nuclide except Cs 137 toward the sediment surface. The more rapid movement of Fe 55 suggests its behavior is produced by different chemical processes than is that of Pu or Am. (Auth) (LKM)

Sr 90; Cs 137; Pu; Am 241; Fe 55

SEA DISPOSAL; RADIONUCLIDE MIGRATION; FALLOUT; FIXATION; PARTICLES; SOLUTES; SEDIMENTS; TRACERS; MARINE SYSTEMS; CHEMICAL PROPERTIES; SEAWATER; CESIUM 137; PLUTONIUM; STRONTIUM 90; AMERICIUM 241; IRON 55; LABORATORY STUDIES; REVIEW;

<231>
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Thermodynamic Considerations Underlying the Migration of Radionuclides in Geomedia: Oklo and Other Examples.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 355-366), 563 pp. (1979)

The prediction of migration or retention of radionuclides in geomedia is one of the major problems associated with radioactive waste disposal. Use of Eh-pH diagrams based on sound thermodynamic data has proven useful for both the Oklo Natural Reactor and for sedimentary uranium deposits. Appropriate diagrams for temperatures in the range from 25 to 200 C allow predictions to be made which are entirely consistent with available measurements. For geomedia where organics are present and for brines, experimental data must be employed to supplement any predictions made on a purely inorganic thermodynamic data base. Thermodynamic treatment of transport of actinides and/or lanthanides in bedded salt deposits is more problematic, yet first experimental results under high Eh conditions indicate a high degree of retardation of the actinides and lanthanides, lesser retardation of Ru, and very little retardation for Cs, Sr, Tc, I, and Sb. Reducing conditions would increase the retardation for Ru, Tc, and Sb; with clays present Cs, Sr, and possibly I should be less mobile. (Auth) (DCW)

One table lists Eh-pH predictions for 30 important elements at Oklo.

WASTES, RADIOACTIVE; WASTE DISPOSAL; RADIONUCLIDE MIGRATION; RADIONUCLIDES; REACTORS, NATURAL; ACTINIDES; LANTHANIDES; THERMODYNAMICS; CHEMISTRY; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; SALT DEPOSITS; REVIEWS; THEORETICAL STUDIES

<232>
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Nuclide Migration Models for Faulted Media.

BNWL-1978; Nuclear Waste Management and Transportation Quarterly Progress Report, October through December 1975, (pp. 6-8), 67 pp. (1976, March)

A comparison of two mathematical models, one for radionuclide migration through "particulate" geologic media and one for migration through "monolithic" media with faults is discussed. The particulate model assumes a cylindrical column of particles on which equilibrium sorption takes place; the monolithic media are modeled the same way, with the fault represented by a cylindrical open column with sorption occuring on the walls. The result of derivations of material balance over differential column distance and differential time for the two models shows that their form is the same; the only

<232> CONT.
Differences are in the numerical value of the dispersion coefficient and in the relationship for calculating the scheme sorption equilibrium constant. This result suggests that the GETOUT computing scheme is applicable to all media in which salt is absent, requiring only properly determined values of water velocity, dispersion coefficient, and sorption equilibrium constant for its application. (LKH)

MODELS; MODELS, MATHEMATICAL; RADIONUCLIDE MIGRATION; ROCKS; SEDIMENTS; GEOLOGIC STRATA; FAULTS; SORPTION; COMPUTER CODES; COMPUTER PROGRAMS; DISPERSION COEFFICIENT; HYDRAULIC VELOCITY; WASTE MANAGEMENT; GROUND WATER; WASTES, RADIOACTIVE; THEORETICAL STUDIES

<233>
Bertholder, N.C., R.O. Cloniger, D.A. Baker, and G. Jansen, Battelle-Pacific Northwest Laboratories, Richland, WA.

Incentives for Partitioning High-Level Waste.

BNWL-1927; 180 pp. (1975, November)

Incentives for separating and eliminating various elements (particularly the transuranics) from radioactive waste prior to final geologic storage were investigated. Exposure pathways to man were defined, and potential radiation doses to an individual living within the region of influence of the underground storage site were calculated. The accumulated high-level waste (i.e., the fission product waste produced by reprocessing spent fuel) from the U.S. nuclear power economy through the year 2000 was the assumed radionuclide source, and western U.S. desert subsoil was the assumed geologic medium. The results of the study showed that for reasonable storage conditions potential incremental radiation doses would be of the same order as, or less than, doses from natural sources. Therefore, for the situations investigated the incentives for special effort to remove any elements, including the transuranics, from high-level waste are vanishingly small, although, incentives do exist for converting high-level calcine into glass. The study required numerous assumptions concerned with the transport of radioactivity from the geologic storage site to man. The assumptions used, on the whole, maximized estimated potential radiation doses, thus maximizing incentives for removing radio-elements from the waste. Incentives were also maximized by assuming that elements removed from the waste could be eliminated from the earth without risk. The conclusion that partitioning incentives are nonexistent--despite assumptions tending to maximize them--is based on comparison of predicted potential radiation doses with routine doses from natural sources. The study found that there are feasible conditions of geologic emplacement where the predicted incremental dose to man is calculated to be as low as one tenth of "background." Although "background" varies with geographic location and consensus standards of allowable incremental dose have not yet been fully established, it was concluded that high-level waste in a glass form can safely be placed in selected geologic media. The methods used in this study can be extended to evaluate any combination of waste type and geologic media at sites that are candidates for final geologic storage. (Auth) (LKH)

PARTITIONING; TRANSURANICS; WASTES, HIGH-LEVEL; STORAGE, GEOLOGIC; ENVIRONMENTAL EXPOSURE PATHWAY; DOSE COMMITMENTS; RADIATION HAZARDS; RADIONUCLIDE MIGRATION; FISSION PRODUCTS; SPENT FUELS; FUEL REPROCESSING; SOILS; CALCINE; VITRIFICATION; GLASS; PREDICTIONS; THEORETICAL STUDIES; RADIONUCLIDES; STATISTICS

<234>
Bertholder, N.C., and C. Defligh-Price, Battelle-Pacific Northwest Laboratories, Richland, WA.

Diffusion of Radionuclides from Geologic Formations.

BNWL-1970; Nuclear Waste Management and Transportation Quarterly Progress Report, October through December 1975, (pp. 5-6), 67 pp. (1976, March)

A model has been developed for radionuclide diffusion through geologic media by molecular diffusion in the liquid phase and in the gas phase, based upon a previous model of transport by moving groundwater. In the new model, convective transport is neglected and the molecular diffusivities of the various chain members are assumed to be different (for gas phase applications). The boundary conditions for impulse and band releases are the same as in the previous model. Solving the model by the Laplace transform technique using a hypothetical set of boundary conditions for impulse release shows that the effects of diffusion, sorption, and radioactive decay will be similar to those in the previous model, except that entrance of radionuclides to the biosphere is later and activities are lower. During the next quarter a computational scheme similar to the GETOUT code will be developed and computations will be made to compare nuclide discharge into the biosphere from this model with that of the previous model. (LKH)

RADIONUCLIDE MIGRATION; MODELS; MODELS, MATHEMATICAL; ROCKS; SEDIMENTS; GEOLOGIC STRATA; DIFFUSION; GROUND WATER; COMPUTER CODES; BIOSPHERE; GASES; SORPTION; RADIOACTIVE DECAY; WASTE MANAGEMENT; WASTES, RADIOACTIVE; THEORETICAL STUDIES

<235>
Cheng, L., V.F. Hodge, and W.A. Lewis, Scripps Institution of Oceanography, La Jolla, CA.

Polonium and Plutonium in an Intertidal Food Chain.

Nature 269(5631):795-796. (1977, October 27)

Fears have been expressed that radioactivity from seaweed exposed to fallout from nuclear testing could be reaching man through the terrestrial food chain. Investigation of polonium and plutonium in kelp, kelp flies, and their predators suggest that, on the basis of current radiation thresholds for human health and safety, such fears are groundless. Samples of cast weed, consisting mainly of MACROCYSTIS PYRIFERA with some EGREGIA LAEVIGATA, PELAGOPHYCUS PORRA, and the marine grasses PHYLLOSPADIX and ZOSTERA, were collected from the beach at La Jolla, CA

<235> CONT.

in June and August 1975. Adult kelp flies mostly FUCILIA NUPITIDIA and F. SEPARATA, along with a few of the associated predatory tiger beetles (CICINDELA sp.), were also collected. Laboratory-raised common houseflies were also obtained for comparative purposes. All samples were dried in an oven at 80 C overnight. Polonium and plutonium were determined in subsamples of 1.0 g and 20.0 g respectively. The flies and beetles were "wet ashed" by heating in a mixture of nitric and perchloric acids. To monitor the chemical recovery, Po 208 and Pu 242 were added as markers. The results showed that the Po content of laboratory-raised houseflies and of the diet on which they were raised as larvae was 10 and 30 pCi/kg dry wt, respectively. The Po content of the kelp flies was 250 pCi/kg dry wt, about half that of the cast weed. Polonium content of the tiger beetles was found to be about half that of the kelp flies (about 160 pCi/kg dry wt). The proportional reductions may be attributable to mechanical or metabolic discrimination on the part of the consumer. Separation of the hard and soft parts of kelp flies demonstrated that essentially all of the Po is concentrated in the cuticles. In ascending the food chain from kelp to kelp flies, the Po/Pu ratio remains essentially unchanged (250 for kelp, 260 for flies). Corresponding values for MACROCYSTIS and for bryozoa removed from their surfaces were 230 and 1000, respectively, while even higher ratios have been found in the internal organs of some marine fishes. Based on the available data, it is calculated that about 0.2% of the Po and 0.2% of the Pu absorbed or fixed to the major seaweeds may enter a terrestrial food chain. (LKH)

Po 210; Pu 239; Pu 240

POLONIUM; PLUTONIUM; ALGAE; INSECTS; INVERTEBRATES; FOOD CHAINS; ENVIRONMENTAL EXPOSURE PATHWAY; BIOLOGICAL TRANSPORT; LABORATORY STUDIES; FIELD STUDIES; FALLOUT; UPTAKE; RADIONUCLIDES

<236>
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Migration Paths for Oklo Reactor Products and Applications to the Problem of Geological Storage of Nuclear Wastes.

IAEA-TC-119/26; Natural Fission Reactors, Proceedings of an IAEA Symposium, Paris, France, December 19, 1977. International Atomic Energy Agency, Vienna, (pp. 693-699). (1978)

In this theoretical discussion, escape of reactor products from the Oklo deposit is determined to proceed in two steps: escape from the uraninite grains, and transport out of the gangue. Escape from the uraninite grains may be by three modes: by fission recoil, by dissolution and recrystallization of the grains, or by diffusion. Escape by fission recoil accounts for the prompt deposition in the gangue of 5-10% of the products; uranium solubility and isotope depletion data indicate that no significant alteration of the uraninite grains took place by dissolution/recrystallization. The rate of loss via diffusion was very slow, reaching a peak during the operating period of the reactor, when it may have been on the order of 10(E-6) fraction/a for the most volatile elements. The quantitative data for Zr, Nb, and Mo indicate that they were wholly retained in the gangue. A small fraction of Tc 99 and a major fraction of the Mo and Pb were lost. Cd may have been lost. Rh, Ag, and Te were significantly retained. On the whole, it appears that elements which were least mobile after leaving the uraninite formed insoluble oxides, silicates or metals, while those which precipitated as sulfides were relatively more mobile. Applying this data to the problem of radwaste disposal, it should be possible to store wastes in a highly stable matrix such as synthetic uraninite in appropriate geologic environments where temperatures can be held below 400 C with loss rates from the matrix under 10(E-6)/a; less volatile elements, such as Pu, should be released at rates less than 10(E-10)/a. Several important assumptions are made: 1) that synthetic uraninite will behave like the natural Oklo uranium mineral; 2) that the early intense radiation field will not lead to loss of the desirable crystalline properties of uraninite; 3) that temperatures of the stored material will not rise to values so high that they lead to a different pattern than at Oklo; and 4) that the geochemical conditions which were responsible for the stability of the Oklo uraninite can be sufficiently specified, satisfactorily reproduced, and assumed to persist for a long period. If these uncertainties can be favorably resolved, synthetic uraninite (or thorianite) may eventually prove useful for the long-term storage of reactor products in geological sites. (Auth) (LKH)

REVIEWS; THEORETICAL STUDIES; URANINITE; MINERALS; RADIONUCLIDE MIGRATION; BRANNERITE; FISSION; FISSION PRODUCTS; DIFFUSION; TECHNETIUM 99; OXIDES; SILICATES; METALS; SULFIDES; WASTE STORAGE; WASTE DISPOSAL; STORAGE, GEOLOGIC; REACTORS, NATURAL; WASTE MANAGEMENT; WASTE-ROCK INTERACTIONS; WASTE TREATMENT

<237>
Dames and Moore, White Plains, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 21. Ground Water Movement and Nuclide Transport.

NUREG/CR-36/21; 335 pp. (1978, April)

This report presents the results of computer analysis of the effects of the construction of a radioactive waste repository and consequent heat generation on ground water movement. A literature search was made to determine typical generic, geologic, and hydrologic properties of the three geomedia under consideration: granite, basalt, and shale. Additionally, the source concentrations and leach rates of selected radionuclides were studied in relation to the estimated ground water inflow rates. Long-term migration of radionuclides as affected by ground water flow was also evaluated. For all rock types under consideration, it is assumed that the proposed repository is at a depth of 1350-2000 ft, with hydrostatic pressures on the order of 40-60 bars, and rock types are of large areal extent, low topographic relief, and have a hydraulic gradient of 1/1000. Ground water flow into shafts built in the generic granite site would be on the order of a few gpm, while flows into shafts

<237> CONT.
in shale and basalt sites would be over 1000 gpm, unless engineering measures are taken to control inflow. If the proposed 6 shafts were constructed simultaneously, mutual interference in water levels would occur, resulting in a reduction of flow to individual shafts by a factor of about 2. During operation of the repository, inflow rates from adjacent aquifers will be on the order of 10's of gpm for granite, 100's of gpm for basalt, and 1000's of gpm for shale. Inflow rates can be reduced by appropriate dewatering wells if required. In all three rock types, heat flux from radioactive decay was significant in distorting the prevalent horizontal ground water flow. Maximum upward flow through the repository is estimated to be the order of 100 gpm for shale, 30 gpm for basalt, and 2 gpm for granite. The shortest estimated travel time for ground water flowing through the repository to a near surface environment is on the order of 50 yrs for basalt and granite and 100 yrs for shale, coinciding with that for the maximum thermal influence, 1000-5000 yrs after repository decommissioning. Estimated surface approach times were on the order of 300 yrs or more in the first 500 yrs after decommissioning. Results of the radionuclide mass transport studies indicate that all nuclides modeled would exit the 3-mile boundary of the site at activities below 10(-10) Ci/yr, except for Tc 99, which appears at the boundary 400-500 yrs after decommissioning and resaturation at a concentration 1000 times acceptable levels. Time did not permit coupling of the ground water and radionuclide migration models with the thermo-mechanical analyses or sufficient interactions among these studies. This could represent a major uncertainty in the suggested allowable thermal loads presented elsewhere. More complete analysis of this problem may show that lower areal thermal loads would reduce or eliminate problems with migrating radionuclides. (Auth) (LFS)

Sr 90; Tc 99; Cs 137; Sm 151; Eu 154; Pu 239; Pu 239; Am 241; Cm 244; Pu 245

COMPUTERS; MODELS; MODELS, MATHEMATICAL; DEPOSITION; STORAGE, GEOLOGIC; SITE SELECTION; RADIONUCLIDE MIGRATION; SHALES; BASALTS; GRANITES; GROUND WATER; LEACHING; HEAT SOURCES; HEAT EXCHANGE; HYDRAULIC GRADIENT; TECHNETIUM; THEORETICAL STUDIES; WASTE STORAGE; WASTE DISPOSAL; WASTES, RADIOACTIVE; RADIONUCLIDES; ROCK MECHANICS; DEFORMATION; CREEP; THERMAL PROPERTIES; [illegible] DEPOSITS; SALTS; WASTES, HIGH-LEVEL, [illegible]ESS

<238>
Davis, W., Jr., R.E. Blanco, R.C. Finney, G.S. Hill, R.E. Moore, and J.P. Witherspoon, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN; Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Environmental Impact.

CONF/NUREG/TM-0; Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing Uranium 233 and Thorium, Section 7.0, (pp. 71-82), 182 pp. (1976, May)

The radiological impact of the model reprocessing plant is assessed by calculating radiation doses to individuals and populations, for each site and relevant treatment case. Release of airborne effluents to the atmosphere is the major pathway for environmental contamination from fuel reprocessing facilities. Atmospheric transport is the principal mode of delivery of radioactive materials to terrestrial environments. Population distributions representative of southeastern coastal and midwestern environments are the average of each distributions around two fuel fabrication plants and one processing plant for each area. The dose conversion factors used in the DOSRAD code to estimate doses resulting from submersion in the airborne effluent, exposure to contaminated ground surface, and intake of radionuclides through inhalation and ingestion are calculated with computer codes which use dosimetric criteria of the International Commission on Radiological Protection and other recognized authorities. Estimates of intake of radionuclides by man through terrestrial food chains were made with a model and computer code, incorporated within the DOSRAD code, which considers transfer of radioactivity to man via ingestion of food crops, beef, and milk. Approximately 9% of the estimated total body dose to individuals living within 55 miles of the model fuel reprocessing plant is the result of external exposure from submersion is air (3%) and exposure to contaminated ground (6%). Internal exposure from inhalation accounts for approximately 5% and ingestion contributes the remaining 86% of the total body dose. Ingestion is the major pathway of exposure to internal organs. In the case of skin, submersion in the airborne effluent, where Kr 85 contributes approximately 100% of the submersion dose, is the major exposure pathway. The relatively high organ dose of 305 millirem/year in one case at the midwestern site is received by the thyroid and is due primarily to H 3 (45%), C 14 (28%), and to a lesser extent to Sr 90 (12%). Also given are estimates of future potential radiation doses to individuals and populations exposed after reprocessing plant has ceased to operate. (ERT)

REACTORS, HIGH-TEMPERATURE GAS-COOLED; RADIATION DOSE; POPULATIONS; WASTE TREATMENT; DOSE COMMITMENTS; CALCULATIONS; EXPOSURE, INTERNAL; EXPOSURE, EXTERNAL; INGESTION; INHALATION; BODY ORGANS; ADULTS; CHILDREN; THYROID; GASES; EFFLUENTS; METEOROLOGY; AIR; WINDS; INGESTION; FOOD CHAINS; RESUSPENSION; SOILS; CARBON DIOXIDE; CARBON 14; CESIUM 134; CESIUM 137; IODINE 129; IODINE 131; KRYPTON 85; TRITIUM; STRONTIUM 90; RUTHENIUM 106; COMPUTER CODES; SKIN; BONES; GASTROINTESTINAL TRACT; THEORETICAL STUDIES; MAN; VEGETATION

<239>
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Dispersion in Flow through Porous Media.

Ground Disposal of Radioactive Wastes, R.J. Hanson (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959, University of California, Berkeley, CA, (pp. 123-130), 160 pp. (1961, July)

The phenomena collectively called dispersion at a fluid-fluid interface are presented.

<239> CONT.

The causes of dispersion are either "macrostructural" or "microstructural". Macrostructural causes occur over several grains and pore spaces so velocity variations in the pores may be neglected. If the medium is homogeneous then the advancing fluid front will be vertical, and if the medium is inhomogeneous the front will interfinger. Equations are included to describe the two processes. When one fluid is more dense than the other, the interface takes on a wedge shape. Instability along the interface is described mathematically, as is the wedge phenomenon, as a sinusoidal-shaped finger. The microstructural causes relate to the velocity differences occurring in the different pores. Molecular diffusion may be assumed to spread the velocities so a turbulent shape will be described. However, applying probability calculus to the distribution may better describe the front. Assuming a tracer is introduced at a point and subsequently moves through an increasing number of pores of the same average velocity, then the probability a single particle chooses a particular channel has an assigned probability. The probability is calculated as the fraction of discharge taking place through the channel, which is numerically equal to the cos (theta). By applying MARKOV's method the probability of particle arrival can be calculated. Molecular diffusion becomes very important for very small velocities and equations are provided in the discussion. (DBV)

DIFFUSION; EQUATIONS; FLUID MECHANICS; HYDRODYNAMICS; THEORETICAL STUDIES; NUMERICAL METHODS

<240>
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Radionuclide Migration Studies Associated with the WIPP Site in Southern New Mexico.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 395-398), 563 pp. (1979)

Results are presented of a study on sorption coefficients (Kd) for 12 radionuclides in natural water and simulated water from various strata in the Delaware Basin. This basin is comprised of bedded salt deposits of which some are being considered for a repository (WIPP). In the natural groundwater Tc and I were not absorbed to any extent, while Cs and Sr exhibited Kd's less than 1. Coefficients for La and Ac were greater than 10 (ml/g) and ranged from 25 to greater than 70 (ml/g) for Ru and Sb. The Kd's in the simulated solutions were higher for the nuclides except Tc and I which remained the same. Studies involved with pH and organic complexants are underway. Another experiment is the assessment of Kd measurement utility. The rock chosen was the Rustler dolomite. For a sample from the 730 ft level the Kd was 10, however from the 789 ft level the Kd was 660. Using a one-dimensional flow equation breakthrough of Cs will occur in 100 yr. Breakthrough experiments are currently being run with I. (DBV)

Distribution Coefficient

ABSORPTION; BASINS, STRUCTURAL; BREAKTHROUGH DISTRIBUTION; DOLOMITE; LABORATORY STUDIES; GROUND WATER; RADIONUCLIDE MIGRATION; SALT DEPOSITS; DISPERSION COEFFICIENT

<241>
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Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico.

SAND-78-0797; 49 pp. (1978, June)

A survey of the potential of geological media from the vicinity of the Waste Isolation Pilot Plant site in southeastern NM for retardation of radionuclide migration via an aqueous carrier has been carried out. Solution simulants representing water in equilibrium with potash minerals of halite and a typical groundwater were spiked with radionuclides and contacted with geological samples to obtain sorption coefficients (Kd's). The samples used were taken from core samples from ERC 9E and ERDA 9 boreholes, and included 8 of halite, 1 of polyhalite, and 4 of the Culebra anhydrite, all from the Salado Formation. Other samples were from the Magenta and Culebra Members of the Rustler Formation, the Bell Canyon Sandstone, and 2 samples from the Cherry Canyon Formation. The nuclides chosen for study were Cs 137, Sr 85, I 131, Tc 99, Sb 125, Co 144, Eu 152, Gd 153, Ru 106, Ac 203, Cs 244, and Pu 238. In the brine simulants, anionic species, TcO_4^-, and I^-, showed little or no tendency to adsorb on any of the geomedia (Kd than or equal to 1), with the possible exception of a clay sample from a halite stratum (Kd less than 5). Cs and Sr Kd's were also generally less than 1, but values in the range of 10-20 were observed on several minerals. Ru Kd's ranged from 25 to more than 1000, depending on the brine and geomedia. Lanthanide and actinide Kd's were typically greater than 1000, with only polyhalite showing significantly lower adsorption. In the groundwater simulant, Tc and I showed the same behavior. Kd's of the other nuclides were slightly higher, particularly those of Cs and Sr (less than 1 to 100). This would be expected if sorption were due to an ion exchange mechanism, but, in the case of the actinides and lanthanides, may also result from an increased contribution of hydrolysis to the Kd due to the higher pH of the groundwater simulant. (Auth) (LEH)

Sorption Coefficient

Cs 137; Sr 85; I 131; Eu 152; Gd 153; Ru 106; Tc 99; Sb 125; Co 144; Ac 203; Cs 244; Pu 238

LABORATORY STUDIES; WIPP; GROUND WATER; BRINES; RADIONUCLIDES; SORPTION; ADSORPTION; ROCKS; SEDIMENTARY ROCKS; HALITE; POLYHALITE; POTASH; EVAPORITES; CLAYS; SANDSTONES; RADIONUCLIDE MIGRATION; SORPTION COEFFICIENT; REPOSITORY; SITE SELECTION; STORAGE, GEOLOGIC; WASTE STORAGE; COPES, JEPPERS; SALT DEPOSITS; ION EXCHANGE; WASTE MANAGEMENT; MINERALS

<202>

ENVIRONMENTAL TRANSPORT

<201>
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Flow in Fractured Porous Media.

Water Resources Research 13(3):558-566.(1977, June)

The equations governing the flow of fluid through fractured porous media are derived. These equations consist of Darcy's law for fluid flow in the primary pores, equations of motion for fluid flow in the fractures, and two continuity equations. The system of equations is coupled by the interaction of fluid in the primary pores with fluid in the fractures. The coupling terms, which are incorporated in the continuity equations, describe the transient flow of fluid out of the primary pores and into the fractures. The finite element Galerkin method is used to solve this coupled system of equations for transient flow in a confined leaky aquifer. Solutions are obtained for both constant discharge and step drawdown problems. The importance of coupling the primary blocks to the fracture system and the effect of the acceleration term in the equation of motion in the fractures are studied by using this model. (Auth)

AQUIFERS; DRAWDOWN; EQUATIONS; FLUID MECHANICS; HYDRODYNAMICS; MODELS, MATHEMATICAL; PERMEABILITY; POROSITY; WELLS; NUMERICAL METHODS; FRACTURES; THEORETICAL STUDIES

<203>
Duguid, J.O., and M. Reeves, Oak Ridge National Laboratory, Oak Ridge, TN.

Material Transport through Porous Media: A Finite-Element Galerkin Model.

ORNL-4928: Environmental Sciences Division Publication 733; 201 pp.(1976, March)

A two-dimensional model to describe the flow of a dissolved constituent through porous media was developed. Mechanisms included in the model are advective transport, hydrodynamic dispersion, chemical adsorption, and radioactive decay. The solution of the equations describing the above mechanisms was found through the use of Galerkin finite-element methods. These methods are ideal for numerical simulation of transport because numerical dispersion is small and multidimensional irregular regions can be easily treated. The major purpose of this model is to allow the investigator the ability to simulate the transport of toxic materials through saturated-unsaturated porous media for prediction of future concentrations in the ground water. The development of the model is outlined in detail and two examples are fully presented. (NDV)

The appendices include the listing of the computer code, input, and output for two examples. A flow chart of the program is included in the text of the report.

COMPUTER CODES; COMPUTER PROGRAMS; EQUATIONS; HYDRODYNAMICS; MODELS, MATHEMATICAL; NUMERICAL METHODS; RADIONUCLIDE MIGRATION; GROUND WATER; THEORETICAL STUDIES

<204>
Erdal, B.R., R.J. Daniels, D.C. Hoffman, F.D. Lawrence, and L. Wolfsberg, Los Alamos Scientific Laboratory, Los Alamos, NM.

Sorption and Migration of Radionuclides in Geologic Media.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 423-426), 562 pp.(1979)

Beneficial utilization of nuclear power requires demonstration to the public that nuclear wastes can be stored with assurance of safety for current and future generations. The LASL Radiochemistry Group is participating in programs to evaluate the potential interactions between radioactive waste, phreatic water, and surrounding rock which might affect the long-term containment of radionuclides in geologic media. Initially we are studying the interactions of material from the Eleana Shale Formation, the Climax Stock Granite, and several Jackass Flats tuffs with various radionuclides in selected ground waters. Petrographic and microautoradiographic techniques have been developed to identify specific sorption sites. Extensive batch and column measurements of sorption characteristics have been made. The effects of specific variables, such as temperature, permeability, porosity, cation exchange capacity, particle size, and surface area, have been investigated. (Auth)

Abstract from conference is complete text.

WASTE-ROCK INTERACTIONS; SHALES; GRANITES; TUFFS; SORPTION; RADIONUCLIDES; PETROGRAPHY; AUTORADIOGRAPHY; METHODS; WASTES, RADIOACTIVE

<205>
Fernandez, S.J., G.D. Pierce, D.C. Wetzer, and R.G. Notes, Idaho National Engineering Laboratory, Idaho Falls, ID.

Methods Evaluation for the Continuous Monitoring of Carbon 14, Krypton 85 and Iodine 129 in Nuclear Fuel Reprocessing and Waste Solidification Off-Gas.

ICP-1187; 28 pp.(1979, March)

The state-of-the-art for the continuous monitoring of C 14, I 129, and Kr 85 released from reprocessing plants and solidificatoin facilities was evaluated. Published methodology and developmental programs at the Idaho Chemical Processing Plant are summarized. The presence of high concentrations of SO2 provides interferences for most methods currently employed. The separation of C 14 and I 129 from the SO2 and Kr 85 in the sample stream is required for accurate monitoring. The method proposed for development consists of the catalytic conversion of the C 14 to CO2 and the I 129 to I2. These species are then separated from possible interferences by selective permeation across thin membranes. The Kr 85 is counted directly with a beta detector in a flow chamber since it does not suffer from significant interferences. The separated C

<285> CONT.
Xe is counted with either a flowthrough proportional counter or an ionization chamber. A laser fluorescence method is proposed to detect the I 129. A developmental program for extending the state-of-the-art is outlined. (Auth) (PAP)

EVALUATION; METHODS; MONITORING; CARBON 14; KRYPTON 85; IODINE 129; IODINE; GASES; REPROCESSING; SOLIDIFICATION; NUCLEAR FACILITIES; RADIATION DETECTORS; BETA PARTICLES; OXIDATION; NITROGEN OXIDES; SEPARATION PROCESSES; CARBON DIOXIDE; SCINTILLATION COUNTERS

<286>
Flynn, K.F., R.E. Barletta, L.J. Jardine, and R.J. Steindler, Argonne National Laboratory, Argonne, IL.

Leach Rate Characterization of Solid Radioactive Waste Forms.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 149-155), 563 pp. (1979)

Leaching characteristics of solid radwaste forms were determined by use of lithium drifted germanium detectors for gamma ray measurements and gas flow proportional counters for beta detection. Waste materials which have been characterized are calcine, sintered calcine granules, borosilicate based glass, composites of sintered stabilized calcine granules in metal matrices, and potential metals for waste matrices. Distilled water, saline solutions, carbonate saturated aqueous and acetate buffered aqueous solutions were the chosen leaching media. Changes in the leach rates for each element studied as a function of temperature, time and leaching media were not great. Problems with this technique have been considered and applications to other types of wastes are being investigated. (NDV)

LABORATORY STUDIES; METHODS; LEACHING; WASTES, SOLID; WASTES, HIGH-LEVEL; CALCINE; GLASS; ENTHES; TEMPERATURE

<287>
Frejacques, C., Commissariat a l'Energie Atomique, France.

Conclusions Drawn from Studies on the Oklo Natural Nuclear Reactor Concerning the Evolution in the Ground of Radioactive Waste over a Long Period.

CONF-750611; Nuclear Energy Maturity, Proceedings of the 1st European Nuclear Society Conference, Paris, France, April 21-25, 1975, (pp. 695-696); Transactions of the American Nuclear Society 20:695-696. (1975)

First results of a localized study of isotopes still present at the site of the Oklo natural reactor show that the rare earths and Pu formed remained in the reactor area. Kr and Xe disappeared almost completely (about 1% of quantities initially formed remain), preventing conclusions about radioiodine evolution. Replacement of the alkaline and alkaline-earth isotopes by natural elements masks information about Cs 135 and Cs 137; the relatively low Zr 96 concentrations may indicate an important migration of Sr 90. Ru, Tc, Ag, Pd, Rh, and the rare earths stayed on-site. Considering the conversion rate (0.45), comparison of fluence and loss in U 235 shows that most of the Pu stayed on-site and reverted to U 235. Conclusions about migration of Np 237, found as Bi 209, are difficult to establish, due to the small initial quantities of Pu 241 formed. Fixation of transuranics and fission products in hydrous clays at Oklo for 2 billion years is reassuring; however, the findings also indicate a need for concern about recovery and fixation of the more long-lived elements: gases, iodine, and the alkaline and alkaline-earths. Transport of Pu as solid powder may be more dangerous than fixation of liquid Pu in the ground, due to possible aerosol formation. (LKS)

LABORATORY STUDIES; GEOLOGIC DEPOSITS; RARE EARTH COMPOUNDS; PLUTONIUM; RADIONUCLIDE MIGRATION; URANIUM; REACTORS; FIXATION; CLAYS; IODINE; FISSION PRODUCTS; RADIONUCLIDES; WASTE MANAGEMENT; WASTE DISPOSAL; WASTES, RADIOACTIVE

<288>
Fried, S., A. Friedman, F. Schreiner, R. Sjoblom, J. Hines, R. Strickert, and P. Pickert, Argonne National Laboratory, Argonne, IL.

Measurements of Parameters Essential to Predicting Nuclide Migration.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 367-377), 563 pp. (1979)

It is known there are a number of thermodynamic parameters which are essential to predictions of nuclide migration in the geosphere. Even the most simple of these, such as the absorption parameters, and identification of the most probable chemical species, have to be determined for each system. In addition, there is a growing awareness that nonequilibrium effects are important and that measurements have to be performed which relate to these kinetic parameters. In this paper examples are presented of these types of measurements and their applications. A series of macroscopic samples of plutonium were made in the +3, +4, or +6 oxidation states and equilibrated with geologically significant media to determine the behavior of the various oxidation states. Measurements of this type are obviously important. As an example of the kinetic studies, the temporal study of a series of absorption curves for Np on crystalline metamorphic rock which was a gray hornblende schist, were measured and correlated with the migrational behavior for this species. As a more complete example of these techniques, the stability of U(+6) was determined in Bandelier tuff, the kinetics of the absorption was measured; and finally the results were used to predict the movement of U(+6) through blocks of tuff. A laboratory migration test of this movement was performed to check the prediction. (Auth) (Complete text)

RADIONUCLIDE MIGRATION; PREDICTIONS; ABSORPTION; SCHISTS; TUFFS; URANIUM; AMERICIUM; PLUTONIUM;

<248>

ENVIRONMENTAL TRANSPORT

<248> CONT.
LABORATORY STUDIES

<249>
Fried, S., A.M. Friedman, R. Atcher, and J. Hines, Argonne National Laboratory, Argonne, IL.

Detention of Plutonium and Americium by Rock.

Science 196:1087-1089 (1977, June 3)

Experiments have been conducted to determine migration and retention of the radionuclides Pu 237, Pu 238, and Am 241 in selected rock types. From these data the migratory behavior of actinides away from disposal sites can be inferred and extrapolations made which allow for catastrophic intrusion of water or seismic activity resulting in fissures. The techniques developed may also serve as a basis for evaluating proposed repository sites. Four sets of experiments were carried out: (1) simulation of the migration of actinides through a fissure, (2) determination of migration through relatively impervious limestone, (3) examination of elution characteristics using a small rock sample as an ion-exchange column, and (4) measurement of actual migration in a scale model involving wet and dry cycles. Important results included: (1) more than 95% of the actinides remained very close to the deposition site, with upper limits of relative migration ratios of 100 um/s for Pu and 500 um/s for Am; (2) once the deposits have dried on rock surfaces they are virtually immobile; (3) in all experiments two forms of Pu were observed, the more rapidly migrating form constituting 1-5% of the total activity and apparently moving 3-5 times faster than "normal" Pu. This movement may be explained chemically or by differences in water flow through pore spaces. These data suggest that radioactive decay will have taken place to such an extent that little plutonium or americium can reach the external environment from a well-designed and isolated geological repository site. (LKH)

Migration Rate

Pu 237; Pu 238; Am 241

ADSORPTION; AMERICIUM; FISSURES; BASALTS; IGNEOUS ROCKS; STORAGE, GEOLOGIC; TRACERS; TUFFS; WASTE STORAGE; WASTES, HIGH-LEVEL; LABORATORY STUDIES; LIMESTONES; POROSITY; PLUTONIUM; RADIONUCLIDE MIGRATION; ACTINIDES; ROCKS, SEDIMENTARY ROCKS; WASTES, RADIOACTIVE; POLYMERS; WASTE-ROCK INTERACTIONS

<250>
Fried, S., A.M. Friedman, D. Cohen, and R.G. Strickert, Argonne National Laboratory, Argonne, IL.

The Migration of Long-Lived Radioactive Processing Wastes in Selected Rocks.

ANL-78-46; 83 pp. (1978, April)

Research is being conducted at Argonne National Laboratory to determine the migration characteristics of long-lived actinides and fission products in the lithosphere. A study is in progress on the behavior of Np 237 in selected rocks and minerals. Molar absorptivities and solubilities of Np(V) as a function of pH are given. Sorption was greatest on apatite, azurite, siderite, limestone, granite, shale, and sandstone. Magnetite and basalt showed some sorption; the sulfide minerals exhibited none at all. It was found that Fe(II) in groundwater will reduce Np(V) to the relatively immobile Np(IV); similar effects on Pu are under investigation. In another study, vertical distribution of actinides in samples of tuff from Los Alamos Scientific Laboratory (LASL) and sand/clay from Chalk River National Laboratory (CRNL) burial grounds were compared with a laboratory model of Pu-Am migration. Pu from the LASL material correlated well, while Am showed greater migration. The difference is believed to be due to uncertainty in the source term. Work on the CRNL material has not been completed. Sorption methods other than ion exchange were examined for the immobilization of Tc and I by various rocks and minerals. Samples tested included tuff from LASL, basalt from Idaho Falls, and rocks from the Nevada Test Site. Results showed that up to 99% of Tc and I tracers are absorbed by certain minerals by mechanisms other than ion-exchange, possibly via redox reactions. Sorption of I and Tc by sulfide minerals is apparently limited, although Cu(I) and Pb sulfides are more effective sorbants than oxides. Iodine absorption may involve coprecipitation or incorporation into crystal lattices. Iodate absorption displays similar features, but the mechanism is uncertain. The rate of absorption is slow (disequilibrium occurs at water front velocities of 10-80 km/yr). Temperature and mineral surface area effects remain to be considered. (LKH)

Percent Adsorption

Np 237; Pu 237; Pu 239; Am 241; Tc 99; I 129; I 131

ACTINIDES; FISSION PRODUCTS; ROCKS; MINERALS; TUFFS; BASALTS; RADIONUCLIDE MIGRATION; IODINE; TECHNETIUM; AMERICIUM; PLUTONIUM; NEPTUNIUM; SULFUR COMPOUNDS; ION EXCHANGE; IMMOBILIZATION

<251>
Fried, S., C. Sabau, J. Hines, and A. Friedman, Argonne National Laboratory, Argonne, IL.

Migration of Uranium Process Wastes from the Uranium 233-Thorium 232 Cycle.

ANL-78-22; 19 pp. (1978, March)

Absorption coefficients and elution behavior of U(VI) in groundwaters have been measured in columns of rhyolitic tuff (Los Alamos tuff) for the purpose of determining parameters influencing the migration of uranium from a waste repository. This has become a problem since August 1977, when the Shippingport reactor, which uses U 233 fuel and a Th 232 blanket, became operational; the fuel system this reactor uses introduces a new aspect to the waste isolation problem - namely the immobilization of U 232 and U 233 (and their daughters) as contrasted with the present problem of disposal of the transuranic wastes (Np, Pu, Am, and Cm) resulting from operation of LWR or plutonium breeders. Three aspects of the rock-uranium reaction were examined in this study. The first was the capacity under saturated flow conditions. The second was migration in an

<251> CONT.

unsaturated rock sample simulating the conditions under which certain repositories have already been operated. The third aspect was to attempt to determine the batch Kd with crushed samples of rock at two pH's (ranging from 4.0 to 5.8, depending on the sample). The first experiment resulted in elution of U 233 from the saturated cylindrical rock sample with peaks at 32, 91, 110 and 116 free column volumes (fcv). The pH of the applied U 233 solution was 5.42 and pH of the tuff-equilibrated-water eluting solution was 5.6. The column capacity was 1.6 X 10(E-3) millimoles U(VI)/g. The experiment was repeated at pH 8.08 (applied) and pH 3.8/(eluate), with resulting peaks at 12, 54, 64, and 78 fcv and a capacity of 7.8 X 10(E-4) millimoles U(VI)/g. A crushed sample treated with a 3.02 pH U solution and eluted at 3.02 pH yielded a capacity of 3.1 X 10(E-3) millimoles U(VI)/g; no definite peaks were observed. In the second experiment, three notches were cut in the top of a block of tuff and a 1 X 10(E+6) d/m U 237 solution was pipetted into each of them; the notches were allowed to dry, then 7 liters of water were applied in 1 liter doses, the rock being allowed to dry between doses. The site of one of the notches was then cored to determine U penetration. The other two sites were cored after doses of 14 and 21 liter of water, respectively. The results showed that 90% of the U remained in the first cm of rock in all cases; about 5% or less of the U migrated further in response to larger quantities of "rain". The reduced migration compared to the saturated situation appears to indicate that allowing the U to dry on the rock permits formation of a more tenacious bond, and that the U is very difficult to remobilize. Results of the batch Kd measurement were ambiguous. Values were 77-178 for six samples, but pH changes during the experiment and the inhomogeneity of the rock make it difficult to judge their validity. The behavior of U(VI) is not the same as Pu(VI) on tuff; the experimental results to date do not reveal the reason for this difference. Kd values in the range obtained imply that geologic containment would be difficult in this rock type. However, it may be possible to find more retentive strata; rocks containing reducing components are the most likely candidates and further investigation is urgently needed if the U 233-Th cycle is to be widely used. (Auth) (LKM)

Sorption Coefficient

U 233; U 237

URANIUM 233; URANIUM 237; TUFFS; IGNEOUS ROCKS; RADIONUCLIDE MIGRATION; ABSORPTION; WASTE-ROCK INTERACTIONS; SATURATED ZONE; UNSATURATED ZONE; SORPTION COEFFICIENT; LABORATORY STUDIES; WASTE MANAGEMENT; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; FUEL CYCLES; REACTORS, LIGHT WATER BREEDER

<252>

Fried, S.M., A.M. Friedman, J.J. Hines, and L.A. Quarterman, Argonne National Laboratory, Argonne, IL.

Annual Report on DOWT PROJECT AN0115A, FY-1975.

ANL-75-68; 14 pp. (1975, September)

Studies in FY 1974 verified that the migration of actinides through a fissure or through pores can be described in terms of a surface absorption coefficient, and that two chemical forms of Pu(IV) are recognizable - one migrating ten times faster than the other. Continuing study in FY 1975 of migration of Am and Pu through an artificial fissure of Idaho basalt showed that Pu activity formed a sharp peak at low fluid flow velocities and a broad peak at high velocities, indicating that the two forms of Pu exchange by a slow equilibrium. The two species of Pu have been separated by elution through plugs of tuff and an HDEHP column (only the ionic species adsorbs). A computer model representing the migration process was developed and tested by lab experiments involving Pu 237 deposited on the surface of a block of Los Alamos tuff and alternately wetted and dried. The model failed to accurately predict the widths and exponential gradients of the two experimental peaks, and is being modified. Study of surfaces-absorption coefficients for Am in various solutions on basalt showed that equilibrium was usually attained within 24 hours, as opposed to several days for Pu. (LKM)

Migration Rate; Sorption Coefficient

Pu 237; Am 241

ADSORPTION; AMERICIUM; FISSURES; BASALTS; IGNEOUS ROCKS; PLUTONIUM; PLUTONIUM ISOTOPES; RADIONUCLIDE MIGRATION; ACTINIDES; POLYMERS; DIFFUSION; TUFFS; MODELS; TRACERS; LABORATORY STUDIES; COMPUTERS; SORPTION COEFFICIENT

<253>

Frigerio, N.A., K.F. Eckerman, and R.W Stove, Argonne National Laboratory, Argonne, IL.

The Argonne Radiological Impact Program (ARIP), Part I. Carcinogenic Hazard from Low-Level, Low-Rate Radiation.

ANL/ES-26 (Part 1); 35 pp. (1973, September)

Various models and predictions of carcinogenic hazard to man from radiation in the environment have been examined and compared with actual experience in U.S. and foreign populations. All of the current models have derived their data from small, selected populations, at high dose rates and generally at high dose levels. They have assumed monotonic extrapolation to zero dose, even though this assumption has always been without observational basis. They have also assumed such extrapolation can be made without consideration of dose rate, and assumption not only without observational basis, but one contradicted by a large body of radiological and toxicological data. On these bases predictions have been made of significant increments in malignant mortality rates due to the radiation background. Extensive statistical analysis of actual populations at risk conducted at Argonne National Laboratory shows not only no increment but an actual decrement, and these predictions are left without observational support. Thus, it appears that one or both of the above assumptions is invalid and that background radiation does not constitute an environmental carcinogen of significance. By the same token, the radiation added by nuclear power plants cannot be a carcinogenic hazard either, since it has the same radiobiological character as the current background, but is much lower in both dose

<253> CONT.
level and rate. (Auth) (LKN)

BACKGROUND RADIATION; RADIATION HAZARDS; POPULATIONS; MAN; STATISTICS; MORTALITY; THEORETICAL STUDIES; NUCLEAR POWER; NUCLEAR FACILITIES; TOXICITY

<254>
Gera, F., Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Geochemical Behavior of Long-Lived Radioactive Wastes.

ORNL/TM-4881; 99 pp.(1975, July)

This report attempts to put the long-term hazard potential of high-level wastes into proper perspective by following the changes in isotopic composition of the waste up to 10 million years after generation. It also includes an extensive review of the available information on the geological and environmental behavior of the long-lived nuclides. The computer program ORIGEN has been used to calculate the activities of the radionuclides contained in high-level waste after various decay times. The critical long-lived fission product is shown to be Tc 99, with 143 Ci/cu m of waste produced by reprocessing of 10 MT of LWR fuel, and 182 Ci/cu m of waste from reprocessing of 10 MT of LMFBR fuel. The hazard associated with the long-lived fission products, which is relatively minor, would become comparable to the hazard of the heavy radionuclides only after extremely long decay times (greater than 10 million years). Considering ingestion as the mode of exposure, the main contribution to the long-term hazard [greater than 10(E+5) yrs] from heavy radionuclides is due to natural radioactive elements of the uranium series, with Ra 226 as the critical nuclide; the neptunium series is next in importance. Based on inhalation hazard, Pu 239 remains the critical nuclide after 100,000 years, but a few tens of thousands of years later the main hazard becomes associated with Th 229. Very long term inhalation hazard [greater than 10(E+6) yrs] is due almost completely to the neptunium series. Mechanisms leading to activity release from a repository that have the greatest probability of occurrence are: 1) invasion of the disposal zone by groundwater, and 2) disinterment of the waste through erosion. Regardless of the mechanism, activity would be transferred to the surface very slowly, and would undergo significant dispersion and dilution in the process. Exposure of human populations would depend upon the biologic availability of the nuclides, dietary habits, and the fraction of activity that would be resuspended. With the partial exception of Ra, which is concentrated in the fruits of certain plants, the heavy nuclides are characterized by very low biologic availability in terrestrial systems. Biologic availability may be somewhat higher in aquatic systems, as indicated by Pu concentration factors as high as 2 X 10(E+4) in North Atlantic SARGASSUM and 10(E+3)-10(E+4) in some edible bivalves. Inhalation hazards are controlled primarily by the efficiency of resuspension mechanisms. Soil erosion by wind is more effective for particles 50-500 um in size, while pulmonary deposition of particles larger than 10 um is negligible. (LKN)

TECHNETIUM 99; PLUTONIUM 239; RADIUM 226; THORIUM 229; HAZARD ANALYSIS; WASTES, RADIOACTIVE; WASTE DISPOSAL; REVIEWS; INHALATION; INGESTION; BIOSPHERE; UPTAKE; FOOD CHAINS; PARTICLES, AIRBORNE; EXPOSURE, INTERNAL; EXPOSURE, POPULATION; WASTES, HIGH-LEVEL; STORAGE, GEOLOGIC; WASTE STORAGE; RADIONUCLIDE MIGRATION; GEOCHEMISTRY; ENVIRONMENTAL EXPOSURE PATHWAY; REACTORS, LIQUID METAL FAST BREEDER; REACTORS, LIGHT WATER; FISSION PRODUCTS; ECOSYSTEMS, AQUATIC; ECOSYSTEMS, TERRESTRIAL; COMPUTER PROGRAMS; RADIOACTIVE DECAY; RADIOACTIVITY; RADIONUCLIDES; TIME FACTOR

<255>
Gibbons, J.H., J.J. Alberts, D.C. Adriano, and K.W. McLeod, Savannah River Laboratory, Aiken, SC.

Critical Pathways of Radionuclides to Man from Agro-Ecosystems.

DOE/SR-0536; 11 pp.(1978, February)

A determination of environmental behavior of radionuclides released from an operating nuclear fuel reprocessing facility is made, using true industrial sources. Free controlled source experiments and by comparing the ratio of Pu 238 content to the total Pu content in the various ecosystem components, it was concluded that greater than 90% of the contamination of the biota is due to direct deposition and resuspension. Three soils were collected, differing in their source and level of contamination, and soil physical and chemical characteristics. Two soils were collected from fields in which previous research involving agricultural species had been conducted. The third soil was from the floodplain of a stream which receives effluents from the same reprocessing area. The floodplain soil is dominated by Pu 238 while the Upper and Lower Fields soils contain largely Pu 239 and Pu 240, followed by Am 241 and Cm 244, with concentrations approximately one order of magnitude less in the Lower Field soil. Fertilizer, lime, and chelating agents were mixed with small volumes of dried homogenized soil. Crops were grown in pots containing this soil and cropped to avoid external contamination. Based on these experiments there appears to be little incorporation of the radioisotopes into plant tissues, especially in potentially edible tissues. Although there are slight differences in uptake rates with treated soils (lime or lime plus chelate), these differences are not consistent with regard to treatment. Preliminary conclusions indicate that very little Pu [CR = 10(E-3) to 10(E-4)] will be taken up by the root pathway from soils which contain as high as 5.28 pCi/g of Pu 238. Although resuspension and direct deposition contribute higher amounts of Pu, it will not be important over the long life of some of the transuranic nuclides. Future research direction includes comparative environmental behavior of Pu and U. Transuranic concentrations in crops are given in tables. Tabularized data are included. (CAJ)

Pu 238

AGRICULTURE; BIOLOGICAL TRANSPORT; BIOTA; CURIUM; ARSENICUS COMPOUNDS; CLAYS; CROPS; DEPOSITION; LABORATORY STUDIES; SOILS; PLUTONIUM; URANIUM; RADIONUCLIDE MIGRATION; RADIOECOLOGY; SEDIMENTS; SOIL TRANSPORT; STREAMS; TRANSURANICS; UPTAKE; WASTES,

<255> CONT.
RADIOACTIVE; WASTES, COMMERCIAL; CHELATES; VEGETATION

<256>
Girardi, F., G. Bertozzi, and A. D'Alessandro, Joint Research Center, Ispra Establishment, Chemistry Division, Ispra, Italy.

Long-Term Risk Assessment of Radioactive Waste Disposal in Geologic Formations.

EUR 5902 EN; 42 pp.(1978)

The methodology studied is based on the assessment of the quantitative value of a system of barriers which may be interposed between waste and man. The barriers considered are: 1) geologic segregation; 2) chemical and physical stability of conditioned waste; 3) interaction with geologic environments (subsoil retention); and 4) discrimination factors in the biosphere. The methodology is applied to idealized test cases based on the following assumptions: Wastes are generated during 30 years of operation in a nuclear park (reprocessing and refabrication) capable of treating 1000 ton/yr of light-water reactor fuel. High-level waste is conditioned as borosilicate glass while low- and medium-level wastes are bitumenized. All waste is disposed off into a salt formation. Transport into the biosphere occurs by ground water, with no delay due to retention on adsorbing media. Distribution in the biosphere occurs according to the terrestrial model given. Under these assumptions data were obtained concerning environmental contamination, its levels, contributing elements, and pathways to man. (RAP)

HAZARD ANALYSIS; WASTE DISPOSAL; METHODS; GEOLOGIC FORMATIONS; NUCLEAR FACILITIES; REPROCESSING; WASTE TREATMENT; REACTORS, LIGHT WATER; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; SILICATES; BITUMENS; SOLIDIFICATION; SALT DEPOSITS; ENVIRONMENTAL EXPOSURE PATHWAY; GROUND WATER; CONTAINMENT; ECOSYSTEMS, TERRESTRIAL; STORAGE, GEOLOGIC; THEORETICAL STUDIES; REPOSITORY; ALPHA PARTICLES; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; ACTINIDES; MODELS, MATHEMATICAL; LEACHING; EQUATIONS; SOILS; PLUTONIUM 239; PLUTONIUM 240; AMERICIUM 241; AMERICIUM 243; METEORITES; FLOODING; EXHUMATION; COMPUTER PROGRAMS; RETARDATION FACTOR; MAN; INGESTION; INHALATION; FOOD; SURFACE WATERS

<257>
Giuffre, M.S., and B. Rosa, The Analytic Sciences Corporation, Reading, MA.

The Effect of Retardation Factors on Radionuclide Migration.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 439-442), 563 pp. (1979)

An estimation of different sorption factors of nuclides in the two- and three-member decay chains on the peak and integrated releases from an aquifer are presented. The reconcentration ratio of radionuclide decay products was estimated for the two-member decay chains (daughter nuclide has both a smaller half-life and sorption retardation factor than the parent) and differential equations were used to describe the one-space transport decay, and sorption. By a series of approximations the flow rate of the daughter is shown to be the ratio of the parent to daughter retardation factors times the parent flow rate at peaks. Numerical results are obtained for the U 234 to Th 230 to Ra 226 and Zr 93 to Nb 93m chains. These techniques are a fast and efficient method for estimating the effect of reconcentration. (NDV)

SORPTION; CONCENTRATION FACTORS; EQUATIONS; RETARDATION FACTOR; RADIONUCLIDE MIGRATION; DAUGHTER PRODUCTS; METHODS; THEORETICAL STUDIES

<258>
Godbee, H.W., A.N. Kibbey, J.G. Moore, and D.S. Joy, Oak Ridge National Laboratory, Oak Ridge, TN.

Application of Mass Transport Theory to the Leaching of Radionuclides from Waste Solids.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978, (p. 30), 77 pp.(1979)

Mass transport equations coupled with data from the field is the best way to model the release of radioactivity from solid wastes over a long period of time. Several expressions based on mass transport theory which assumes that diffusion through the solid is the only, or rate limiting process have been used to interpret the leaching of waste solids. These expressions generally fit leach data best over the midrange of time in experiments that have been carried out for sufficient time to establish long-range trends. In an effort to explain some of the observed deviations in leach data from a diffusion only model, expressions are derived which take into account diffusion plus one of the following other mechanisms: 1) a surface concentration of the mobile species which is time dependent; 2) an initial concentration of the mobile species which is a function of position in the specimen; 3) dissolution and precipitation of the mobile species and; 4) dissolution of the matrix material. (Auth) (NDV)

This paper was presented at the conference but was not published in the Proceedings.

MODELS, MATHEMATICAL; DIFFUSION; LEACHING; EQUATIONS; WASTES, SOLID; WASTES, RADIOACTIVE; ENVIRONMENT; THEORETICAL STUDIES

<259>
Goldber, M.H., S.E. Logan, and H.C. Brebanc, U.S. Environmental Protection Agency, Washington DC; University of New Mexico, Albuquerque, NM.

An Assessment Methodology of Environmental Risks Associated with Radioactive Waste Management.

IAEA-CN-36/28; Nuclear Power and Its Fuel Cycle, Proceedings of an International Conference, Salzburg, Austria, May 2-13, 1977.

<259>

ENVIRONMENTAL TRANSPORT

<259> COBY.
International Atomic Energy Agency, Vienna. (10 pp.). (1977, May)

The paper presents an assessment model for the probability of various geological, meteorological, and hydrological events as potential mechanisms for release of radioactive material to the environment. The Environmental Protection Agency, in collaboration with the University of New Mexico is developing this assessment methodology to perform parametric risk calculations for a variety of geologic disposal concepts, fixation processes and reprocessing operations to be implemented for portions of an Environmental Impact Statement (EIS) for assessing possible accidents, potential environmental impacts of man-related and natural faults at storage and disposal sites, and potential release mechanisms and pathways to man. This model (computer code ARHAW for Assessment Method for Radioactive Waste) provides a valuable insight into a typical full-scale repository for all commercial radioactive high-level waste. This probabilistic method assigns a far greater magnitude [by a factor of 10(E+5)] to geologic events which result in an initial release to ground and surface water than events, such as volcanogenic transport, which result in major initial releases to the air receptor portion of the model. Indications are that a major barrier to the release of radionuclides is the low leachable form of the borosilicate glass as well as the accompanying slow dissolution rate of the bedded salt in the Los Medanos area. Preliminary results have shown only Tc 99, I 129, and C 14 exhibit appreciable transport but with values less than 1 x 10(E-22) uCi/sq cm. All other nuclides are sorbed or disappear by decay. Further, results of the potential total economic damages for the full-scale repository at the Los Medanos site show fission products and activation products, in the initial repository, comprising less than 1% of the long-term costs, compared with the substantial contribution of the actinides for the period extending 10(E+6) years. (CAJ)

MODELS; RADIONUCLIDE MIGRATION; ENVIRONMENT; REPOSITORY; STORAGE, GEOLOGIC; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; EVALUATION; ENVIRONMENTAL IMPACT STATEMENTS; METEOROLOGY; HYDROLOGY; GEOLOGY; ACCIDENTS; COMPUTER CODES; WATER; ACTINIDES; FISSION PRODUCTS; ACTIVATION PRODUCTS; ECONOMICS; GLASS; SALT DEPOSITS; FIXATION; TECHNETIUM 99; CAR " 14; IODINE 129; THEORETICAL STUD

<260>
Grimwood, P., N. Hill, and G. Webb, National Radiological Protection Board, England.

Modelling Studies Used to Evaluate Waste Disposal Options.

Nuclear Engineering International 23(266):55-57. (1978, January)

Modelling studies were conducted to evaluate disposal of vitrified high-level waste from reprocessing of nuclear fuels in geologic media and the deep ocean. For deep ocean disposal, two types of containers were considered - stainless steel, with a mean life of a few years, and a hypothetical container with a mean life of a thousand years. Waste leach rate was assumed to be 10(E-5) g/sq cm/day at 4 degrees C. A short-term plume model and a long-term compartment model were used in assessing the waste concentrations as a function of distance and time after disposal. Quantity of waste dealt with is that generated by postulated nuclear power programs to 2000 AD, equivalent to 12,000 GW(e)y. The most important pathway to man was found to be consumption of seafood and food derived from plankton. The most realistic estimates of doses appear to be those calculated for consumption of pelagic fish; peak annual collective dose by this route would be on the order of 10(E+4) man rem. Doses from consumption of plankton-derived food are considered to have been overestimated. Long-lived containment reduces the effects of the shorter-lived radionuclides, but has no significant effect on the others, such as Pu 239 and Ra 226. The geologic disposal model was based on emplacement of all solidified high-level wastes expected to be generated in the UK by 2000 AD (330 GW(e)y) into a matrix of boreholes in a repository 1 km below the surface. Most of the parameters required no specific rock type, but where specific assumptions were necessary, a repository in igneous rock was assumed. Ground water transport is presumed to be the main pathway to man. Containers were assumed to remain intact 1000 years, waste to be in fist-sized pieces, and leaching at a rate of 10(E+5) g/sq cm/day. Taking into account convection, dispersion, sorption, and radioactive decay, peak doses to man from drinking water were found to be 10(E-1)-10(E-2) of ICRP recommended dose limits for individuals for Tc 99 and Np 237, and less than 10(E-3) of the dose limit for the other nuclides (less than 10(E-10) for Pu 239). Collective annual doses are calculated to be less than 10(E+4) man rem. The main result of these studies has been to indicate areas where further research needs to be done. (LEH)

MODELS, MATHEMATICAL; MODELS; SEA DISPOSAL; STORAGE, GEOLOGIC; DOSE COMMITMENTS; WASTES, HIGH-LEVEL; WASTES, SOLID; ENVIRONMENTAL EXPOSURE PATHWAY; CONTAINERS; DOSE RATE; FOODS; SORPTION; THEORETICAL STUDIES

<261>
Grimwood, P.D., and G.A.M. Webb, National Radiological Protection Board, Harwell, Didcot, Oxon, England.

Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste on the Ocean Floor.

NRPB-R-78; 117 pp. (1976, October)

This study is a preliminary assessment of the potential radiological consequences of disposal of solidified high-level radioactive waste on the floor of the deep ocean. As an input to the modelling used in the assessment, an arbitrary choice is made to consider the total high-level waste which would be generated by a postulated world nuclear power program to the year 2000. It is assumed that all this waste, in solidified form, is disposed of on to the floor of the North Atlantic. The body of this report is the modelling of the subsequent release of activity into the water, its dispersion in the ocean and eventual uptake in marine organisms and sediments. The consequent radiation exposure of man is assessed in

<261> CONT.
terms of both individual and collective doses. It is intended that only broad conclusions should be drawn from this study. The objective of the assessment is to highlight those subject areas where more study or information is required before a decision can be reached regarding this method of disposal. No overriding reason connected with the radiological protection considerations has been identified which would preclude the disposal of suitably conditioned high-level waste on the ocean floor. Further evaluation of this disposal option is therefore justified. (Auth)

Comprehensive tables are given of all parameters discussed in the report.

HAZARD ANALYSIS; WASTE DISPOSAL; WASTES, HIGH-LEVEL; SEA DISPOSAL; SOLIDIFICATION; MODELS; REVIEWS; UPTAKE; THEORETICAL STUDIES; FISH; MARINE SYSTEMS; SEDIMENTS; EXPOSURE, POPULATION; MAN; DISTRIBUTION COEFFICIENT; WASTE VOLUME; VITRIFICATION; THERMAL PROPERTIES; LEACHING; CONTAINMENT; SITE SELECTION; CONTAINERS; TRANSPORTATION; SORPTION; RECONCENTRATION; DAUGHTER PRODUCTS; RADIONUCLIDES; INGESTION; RADIATION DOSE; CRITICAL PATHWAY; FISH; MOLLUSKS; CRUSTACEANS; PLANKTON; DRINKING WATER; FOOD CHAINS; TRONION; SURFACE WATERS; CONCENTRATION FACTORS; RESUSPENSION; FALLOUT; FISSION PRODUCTS; AMERICIUM 241; CESIUM 137; PLUTONIUM 239; TIN 126; RADIUM 226; CERIUM; ACTINIUM; ECOSYSTEMS, AQUATIC

<262>
Hagyghion, H., Aktiebolaget Atomenergi, Stockholm, Sweden.

Calculations of Nuclide Migration in Rock and Porous Media, Penetrated by Water.

AB-RP-77-3260; 68 pp. (1977, September 18)

Some physical and mathematical models are given for migration of nuclides in rock and porous media penetrated by water. The cases considered are thermal convection due to the decay heat from radioactive sources and transport due to the hydraulic gradient connected with the geographic structure. The model for thermal convection is highly simplified but is conservative compared to the often made adiabatic assumption which limits convection effects to a region characterized by a high ratio between buoyant and viscous forces. The piezometric head and corresponding gradients are calculated by analytic methods. It is shown that the solutions are strongly dependent upon the variation of the permeability with depth. The special features of migration in cracks are studied. A computer program, MINUTE, was developed for numerical calculations. Times for transport of some important nuclides to the ground surface were calculated using appropriate input data for representative Swedish deposition sites. Error margins are discussed. (Auth)

THEORETICAL STUDIES; RADIONUCLIDE MIGRATION; ROCKS; REPOSITORY; MODELS, MATHEMATICAL; PERMEABILITY; POROSITY; DEPTH; THERMAL PROPERTIES; CONVECTION; TIME FACTOR; DISTRIBUTION COEFFICIENT; RETARDATION FACTOR; FRACTURES; LEACHING; GROUND WATER; DENSITY; EQUATIONS; COMPUTER PROGRAMS; TOPOGRAPHY

<263>
Maastra, , Netherlands Energy Research Foundation, Petten N.H., The Netherlands.

The Response of Salt Migration to Dissolution in Salt Domes and How It Affects a Radioactive Waste Isolation Facility Design.

CONF-761175; Salt Dome Utilization and Environmental Considerations, J.D. Martinez and R.L. Thoms (Eds.), Proceedings of a Symposium, Baton Rouge, LA, November 22-24, 1976. The Institute for Environmental Studies, (pp. 76-92). (1977)

The major potential failure mechanism for long-term isolation of radioactive wastes is that ground water will contact the rock salt containment shield and that a continuous dissolving action will ultimately result in radionuclide migration back into the biosphere. When dealing with assessment models for a containment breach, the assumption is commonly made that the ground water will find its way through the containment wall into the waste burial area. This paper provides evidence that a salt dome which is still in contact with its mother salt layer still, under certain conditions, respond to a continuous ground-water attack by a salt migration towards the surface area where the dissolving action takes place. The driving force behind this salt migration will be the buoyant force of the sediments overlying and surrounding the dome structure. Several parameters that contribute to promote salt flow will increase with the depth, and thus the extent of this mass transport of rock salt in the solid state will increase accordingly as the dissolving action takes place at a greater depth. However, the mass transport of dissolved rock salt into an aquifer will decrease with the depth, because the permeability of that aquifer will decrease with the depth. Therefore, there will be a break-even point in depth beyond which a salt dome will maintain its placement shape despite the continuing dissolving actions. A model approach indicates that it is realistic to assume that this break-even point will be situated at an attainable depth. Model calculations also indicate that the salt migration, as a response to dissolving actions, should be considered as one of starting points for the radioactive waste isolation facility design, such that if this salt migration should prove to be a containment failure mechanism, it only will affect the facility to a limited extent and after a very long period of time. (Auth) (RAF)

SALT DOMES; GROUND WATER; SEDIMENTS; AQUIFERS; PERMEABILITY; DEPTH; MODELS; REPOSITORY; DESIGN; WASTE DISPOSAL; WASTES, HIGH-LEVEL; EVAPORITES; HALITE; DENSITY; THERMAL PROPERTIES; DISSOLUTION; THEORETICAL STUDIES; WASTES, SOLID; DIFFUSION; HYDRAULIC VELOCITY; CONTAINMENT

<264>
Hill, M.D., and P.B. Grimwood, National Radiological Protection Board, Harwell, Didcot, Oxon, England.

Preliminary Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste in Geologic Formations.

<264>

ENVIRONMENTAL TRANSPORT

<264> CONT.
NRPB-R-69; 75 pp. (1978, January)

The purpose of this study is to carry out a preliminary assessment of the potential radiological consequences of disposing of vitrified high-level radioactive waste in geologic formations. The events which could lead to the release of radioactivity from a geologic repository are reviewed and ingress of groundwater is identified as the principal mechanism by which radioactivity may be transported back to the biosphere. A mathematical model of radionuclide migration with groundwater is used to predict possible rates of release of radioactivity into fresh water from a hypothetical repository containing all the high-level waste which may be generated in the UK up to the year 2000. The individual and collective doses which could be received as a result of man's use of contaminated fresh water are evaluated. The numerical results of the study depend very much on the assumptions made and cannot be used to draw any detailed conclusions. The main result is the identification of areas where further studies are required in order to carry out a full evaluation of this disposal option. (Auth)

Extensive tables are given of discussed parameters. Illustrations show disposal concepts, and leaching and discharge rates of selected radionuclides.

REVIEWS; ECOSYSTEMS, TERRESTRIAL; ECOSYSTEMS, AQUATIC; WASTES, HIGH-LEVEL; WASTE DISPOSAL; STORAGE, GEOLOGIC; HAZARD ANALYSIS; VITRIFICATION; SOLIDIFICATION; GROUND WATER; RADIONUCLIDE MIGRATION; THEORETICAL STUDIES; MODELS, MATHEMATICAL; REPOSITORY; GEOLOGIC HISTORY; RADIATION DOSE; SURFACE WATERS; SEISMOLOGY; ROCKS; FAULTS; CONTAINMENT; METEORITES; VOLCANOES; EROSION; HYDROLOGY; WASTE VOLUME; RADIATION EFFECTS; SUBSIDENCE; THERMAL PROPERTIES; DRILLING; SABOTAGE; LEACHING; CONTAINERS; MAN; EQUATIONS; EXPOSURE, POPULATION; SITE SELECTION; LAKES; DRINKING WATER; SOIL TRANSPORT; SORPTION; INGESTION; CRITICAL PATHWAY; TECHNETIUM 99; CESIUM 135; RADIUM 226; SELENIUM 79; ZIRCONIUM 93; IODINE 129; CURIUM 245; NEPTUNIUM 237; URANIUM 233; THORIUM 229; PLUTONIUM 241; AMERICIUM 241

<265>
Nonstead, J.P., Hanford Laboratories Operation, Richland, WA.

The Movement of Liquid through Soils.

Ground Disposal of Radioactive Wastes, W. J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 181-185), 168 pp. (1961, July)

A general discussion of fluid flow in the saturated and unsaturated zones, related to radionuclide retention in the soil is presented. The Poiseuille equation was used to derive the mathematical relationship for unsaturated flow. Mention is made of decontamination capacity of the soil, and suggests that abandonment of the disposal site may result in continued drainage for some time. Evaluation of drainage capacity of the site is important in terms of monitoring requirements of an abandoned disposal site. Laboratory studies have been conducted on drainage characteristics of the soil. By saturating the soil sample and then centrifuging, many decades of simple field drainage can be simulated in a matter of minutes. Centrifuging local soils at 1,000 gravities for one hour corresponds to approximately 114 years at one gravity. The natural sediments were saturated at time of deposition, (5,000 - 10,000 years ago,. The moisture content of these soils (moisture by volume) varied from less than 1% to 10% in an isolated clay lens. It is believed that future disposal sites will be selected on the basis of simple hydrology, deep water table, massive soil profile in the unsaturated zone, and a climate in which evaporation exceeds precipitation to assure little leaching of contaminants. (PTO) (NDW)

Moisture Content

SOILS; DRAINAGE; EQUATIONS; WASTE DISPOSAL; SITE SELECTION; SITE EVALUATION; LABORATORY STUDIES; IMMOBILIZATION; RADIONUCLIDE MIGRATION; HYDRODYNAMICS; GROUND WATER; SATURATED ZONE; UNSATURATED ZONE; DECONTAMINATION; CLAYS; DISPOSAL SITE; SOIL TRANSPORT

<266>
Inoue, T., and W.J. Kaufman, University of California, Sanitary Engineering Research Laboratory, Berkeley, CA.

Dispersion Relations in Injection Disposal of Radioactive Wastes.

Ground Disposal of Radioactive Wastes, W.J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 131-134), 168 pp. (1961, July)

The movement of an injected solvent in three idealized geometries is discussed. The first is a one dimensional system, such as a ditch, in which the waste front would arrive at a given point in a manner that could be described by a step function. For the radial-flow system the front again would arrive at a given point as a step function. In the two well system, one a source the other a sink, the waste would arrive in gradually increasing concentration. These three geometries were considered without dispersion and are covered in mathematical detail. A discussion follows presenting the equations which describe dispersion. (NDW)

DIFFUSION; EQUATIONS; HYDRODYNAMICS; NUMERICAL METHODS; THEORETICAL STUDIES

<267>
INTERCORP Resource Development and Engineering, Incorporated, Houston, TX.

A Model for Calculating Effects of Liquid Waste Disposal in Deep Saline Aquifers. Part 1 -- Development, Part 2 -- Documentation.

PB-256 903; DOI/BP-76/009a; USGS/WRD/WRI-76/056; 263 pp. (1976, June)

A three-dimensional transient mathematical model has been developed to simulate injection of liquid industrial wastes into confined underground saline aquifers. The model consists of two parts: a wellbore model and an aquifer model. The user may specify surface pressure and temperature in

<267> CONT.
an injection well and the wellbore model permits calculation of the pressure and temperature changes in the wellbore. The aquifer model is a numerical finite-difference solution of three basic partial differential equations for the conservation of mass, energy, and contaminant. Velocities (or mass flow rates) are expressed in terms of spatial pressure gradients through the use of Darcy's law for flow in a porous medium, leading to a set of three coupled partial differential equations. The independent variables are the spatial coordinates and time and the dependent variables are pressure, temperature, and concentration. The equations are coupled through the fluid density which is a function of pressure, temperature, and concentration. The model permits the user to specify initial conditions and a number of boundary conditions along the edges of the numerical grid system. In addition, the user may specify heterogeneities in the aquifer porosity, permeability, and specific heat. The net dispersion of heat and mass is expressed as the sum of hydrodynamic dispersion and molecular conduction (or diffusion). The model contains several finite-difference approximations in space and time, and two different methods of solution. This gives the user an option to select the best solution technique for the problem. The model includes the options to compare the simulated pressure, temperature, or concentration values against the observed values and to obtain two-dimensional contour maps of the calculated results. The computer programs are organized to minimize computer time and storage requirements. (Auth) (LBM)

MODELS; MODELS, MATHEMATICAL; AQUIFERS; AQUIFERS, CONFINED; BRINES; WELLS, INJECTION; WASTES, LIQUID; WASTES, COMMERCIAL; CONTAMINANT TRANSPORT; PETROLEUM; HYDRAULIC VELOCITY; HYDRAULIC GRADIENT; TEMPERATURE; CONCENTRATION; EQUATIONS; FLUID MECHANICS; POROSITY; PERMEABILITY; HEAT TRANSFER; DIFFUSION; COMPUTER PROGRAMS; COMPUTER CODES; THEORETICAL STUDIES

<268>
Isaacson, R.E., and D.J. Brown, Rockwell International, Atomics International Division, Richland, WA.

Environmental Assessment Related to Hanford Radioactive Waste Burial.

RHO-SA-[illegible]; CONF-780316; Waste Management and Fuel Cycles 1978, [illegible] Post and M.E. Wacks (Eds.), Proceedings of a Symposium, Tucson, AZ, March 5-8, 1978. Arizona Board of Regents, (39 pp.). (1978, February)

About 190,000 cu m of solid wastes contaminated with two million curies of radioactive fission products and transuranic isotopes have been buried at Hanford since the project started in 1943. Burial practices and site characteristics that are important to isolating radioactive wastes from the biosphere are summarized in this article. Also discussed are ecological aspects and food web relationships at Hanford. The routine monitoring program for detecting migration of radioactive materials at burial grounds is limited to aboveground surveillance. Criteria call for radiation levels to be less than one millirad per hour, and no detectable surface contamination. All burial grounds are surveyed at least semiannually and corrective action is taken when criteria are exceeded. Burial ground problems include: 1) backfill cave-ins due to deterioration of wooden boxes used for defunct industrial equipment; 2) wind erosion and reduction of ground cover; 3) burrowing animals; and 4) deep-rooted plants which bring radioactivity to the surface. Assessments to date indicate that burial practices have effectively prevented environmental contamination, and that the potential dissolution and transport of radionuclides from buried wastes by rainfall or snowmelt at Hanford because vanishingly small. (PAF)

Ion Exchange Capacity; Transmissivity; Hydraulic Velocity

BURIAL; WASTES, SOLID; WASTES, TRANSURANIC; PLUTONIUM; URANIUM; THORIUM; FISSION PRODUCTS; SITE SURVEILLANCE; RADIONUCLIDE MIGRATION; PRECIPITATION, METEOROLOGICAL; METEOROLOGY; GEOLOGY; ECOLOGY; FOOD CHAINS; MONITORING; HYDRAULICS; AQUIFERS; FLOODING; EARTHQUAKES; ROOTS; FIRES; WINDS; EROSION; STABILIZATION, VEGETATIVE; STABILIZATION, PHYSICAL; ANIMALS, BURROWING; GROUND WATER; SOILS; ISOTOPES; GRASSES; STRONTIUM; CESIUM; [illegible]; RARE EARTH COMPOUNDS; SORPTION; DISPOSAL SITE; VEGETATION

<269>
Isherwood, D., Lawrence Livermore Laboratory, Livermore, CA.

Geochemistry and Radionuclide Migration.

UCRL-80[illegible]; Waste Management and Fuel Cycles 1978, R.G. Post and M.E. Wacks (Eds.), Proceedings of a Symposium, Tucson, AZ, March 6-8, 1978. Arizona Board of Regents. (1978, March 6)

A scenario is presented of a geochemical barrier to achieve effective isolation of nuclear waste in a deep geologic depository. The geochemical barrier is identified as a series of waste/water/rock interactions involving sorption (ion exchange), ion filtration, hydrolysis, precipitation, complexing, and leach resistance of the waste form. Theoretically, the geochemical barrier can provide a major line of defense in protecting the biosphere from the hazards of nuclear waste. Preliminary investigations using computer modeling techniques suggest that retardation is an effective control on radionuclide concentrations. Ion exchange reactions slow radionuclide migration and allow more time for radioactive decay and dispersion. For some radionuclides, solubility alone may limit concentrations to less than the maximum permissible now considered acceptable by the federal government. The level of uncertainty in supporting data remains high. The present emphasis on laboratory measurements of Kd's is not the answer. A relationship between laboratory experiments and in situ values for retardation needs to be established. The effectiveness of the geochemical barrier is ultimately related to the repository site characteristics. In site selection, consideration should be given to rock barriers with high ion exchange capacity that might also act as semi-permeable membranes. Also important are the oxidation potential, pH and salinity of the groundwater. (PAF)

THEORETICAL STUDIES; GEOCHEMISTRY; WASTES,

<269>

ENVIRONMENTAL TRANSPORT

<269> CONT.
RADIOACTIVE; STORAGE, GEOLOGIC; SORPTION; ION EXCHANGE; FILTRATION; HYDROLYSIS; PRECIPITATION, CHEMICAL; COMPLEXES; LEACHING; GROUND WATER; SOLUBILITY; RADIONUCLIDES; pH; Eh; REDOX REACTIONS; ROCKS; SOILS; FRACTURES; CATIONS; ANIONS; MINERALS; SHALES; SEDIMENTARY ROCKS; CLAYS; DISTRIBUTION COEFFICIENT; MODELS; COMPUTER PROGRAMS; RETARDATION FACTOR; DENSITY; POROSITY; PLUTONIUM 239; TECHNETIUM 99; IODINE 129; TEMPERATURE; COMPACTION; AMERICIUM 241

<270>
Jakubick, A.T., Kernforschungszentrum, Karlsruhe, German Federal Republic.

Analysis of Plutonium-Release Consequences on the Environmental Geochemistry.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 427-434), 563 pp. (1979)

Industrial handling of plutonium produces small amounts of Pu in the aerosol form. These Pu particles distribute themselves in the same pattern as Pu in fallout, so the soil can be used to determine the Pu pattern around Pu-handling facilities. Plutonium depth distribution was conducted on 7 undisturbed sites, with soil types ranging from quartz sandy to clay-organic. Soils containing illite, montmorillonite, or kaolinite were given special attention. Using 50 samples the downward migration rates were between 0.1 to 1.0 m/m. The Pu transfer response in a partially saturated medium was on the order of 10(E-3). Since this result does not agree with classical migration theory the transfer is a nonchromatographic process. From further study and computer simulation the main variables determining migration are surface exchange capacity, bulk density, and porosity. Fallout measurements have been made and the maximum background level for SW Germany is known. Hence local deviations can be noted from lognormal plots, and through a variety relationships the biologically active top soil can be calculated. This allows for reliable control limits for Pu discharge to be set for a site. (NDV)

AEROSOLS; DISTRIBUTION; ENVIRONMENTAL EXPOSURE PATHWAY; ILLITE; KAOLINITE; MONTMORILLONITE; SOILS; LABORATORY STUDIES; COMPUTER CODES; PLUTONIUM; RADIONUCLIDE MIGRATION

<271>
Birkhan, R.H., D.C. Adriano, and J.C. Corey, Oklahoma State University, Stillwater, OK; Savannah River Laboratory, Aiken, SC.

Comparison of Plutonium Concentrations in Deer from the Southeastern United States and in Deer from an Integrated Nuclear Fuel Cycle Facility.

Health Physics 36:516-519. (1979, April)

The Pu contents of white-tailed deer resident of the Savannah River Plant (SRP) are compared with deer in neighboring locations and the dose-to-man is calculated from ingestion of the deer meat from approximately 1000 deer harvested each year during organized hunts. Preliminary results show that the total Pu concentrations in deer appeared to decrease in the order of bone > or = liver > or = lungs > or = muscle. Compared with deer from Alabama, Georgia, and North Carolina, the bones and livers of deer at SRP have significantly higher Pu contents, while the lungs and muscles do not. Although the differences are not statistically significant, the alpha 3 Pu 238 values for SRP deer are, in general, higher than values for offplant deer. These higher values reflect the contribution of the chemical separation facilities at SRP to the Pu 238 body burden of deer on the site. (NAP)

Pu 239, 240; Pu 238

FIELD STUDIES; PLUTONIUM 239; PLUTONIUM 240; PLUTONIUM 238; ALPHA PARTICLES; ANIMALS; DEER; UPTAKE; NUCLEAR FACILITIES; RADIATION DOSE; MAN; CALCULATIONS; INGESTION; BONES; LIVER; LUNGS; MUSCLE; BODY BURDEN; DOSE CONSTITUENTS

<272>
Klechkovsky, V.M. (Ed.), and G.N. Tsvelishcheva, Academy of Sciences of the USSR, Moscow.

The Behavior of Radioactive Fission Products in Soils.

AEC-TR-2867; On the Behavior of Radioactive Fission Products in Soil, their Absorption by Plants and their Accumulation in Crops, V.M. Klechkovsky (Ed.), (pp. 7-102), 227 pp. (1957, February)

Sorption and elution studies were among the methods used to study the behavior of fission products in soils. The soils for the study were turf-podzol and black-earth soil from the Kamennaya steppe. It was found that the concentration of the radioactive components had almost no bearing on the relative distribution of activity in the sorbing column of the turf-podzol soil. The relative activity along the column remains constant when the solution is repeatedly passed through the column. About 80 to 85% of the total quantity of radioactive substances introduced was sorbed in the first 1 cm section. In the upper 5 to 6 cm section about 99% of the activity is held. From elution studies the turf-podzol soil holds part of the fission products in exchange-ion form. The other fission products can only be moved by an acid reaction. The distribution curve showed a smaller movement and this may result from different reaction and degree of saturation of these soils with bases. Further studies showed that the acidity of the solution is more important than the total concentration of non-radioactive salts. The theory of the sorption processes is broadly outlined. Sorption and elution tests were also run using Sr 90, Sr 89, Y 90, Cs 137, Zr 95, Nb 95, Ru 106, and Rh 106. (NDV)

ABSORPTION; ISOTOPES; FISSION PRODUCTS; ION EXCHANGE; LABORATORY STUDIES; BEHAVIOUR; DISTRIBUTION; RADIONUCLIDE MIGRATION; ROCK-FLUID INTERACTIONS; SOILS; WASTES, RADIOACTIVE

<273>
Knapp, R.R., and R.L. Podio, University of Texas, Austin, TX.

<273> CONT.
Investigation of Salt Transport in Vertical Boreholes and Brine Invasion into Fresh Water Aquifer. 1. Numerical Study of Convective-Diffusive Flow in Wellbores.

CONF-770383; T/OWI/SUB-77/16537; Salt Dissolution, Proceedings of the National Waste Terminal Storage Program Review Meeting, Austin, TX, March 29-30, 1977, (2 pp.), 38 pp. (1977, March)

A conditionally stable semi-implicit finite difference scheme has been developed for simulating natural convection-diffusion processes in rectangular cavities at high Grashof numbers. These numbers may be as high as 1 X 10(E+10). The time step required for stability decreases with increasing Grashof numbers and length to width ratios. Apparently the time step is independent of the Schmidt number. With increasing Schmidt number and decreasing length to width ratios the dimensionless steady-state time increases. The dependence of salt transport evidenced by the Nusselt number is strongly dependent on the Grashof number. Since the salt concentration is essentially uniform in the column at any given horizontal plane the transport problem is one dimensional for the vertical wellbore case. (MDV)

BOREHOLES; CONVECTION; DIFFUSION; CAVITIES; NUMERICAL METHODS; PREDICTIONS; SALTS; MODELS, MATHEMATICAL; THEORETICAL STUDIES

<274>
Knapp, R.R., and A.L. Podio, University of Texas, Austin, TX.

Investigation of Salt Transport in Vertical Boreholes and Brine Invasion into Fresh Water Aquifer. 2. Experimental Study of Convective-Diffusive Flow in Wellbores.

CONF-770383; T/OWI/SUB-77/16537; Salt Dissolution, Proceedings of the National Waste Terminal Storage Program Review Meeting, Austin, TX, March 29-30, 1977, (4 pp.), 38 pp. (1977, March)

Experiments were run to study salt transport in vertical wellbores. The first long term experiment indicates that a condition of near saturation in the lower part of the wellbore is achieved in about 10 days. During this time a total of 32,200 g of salt were dissolved and accumulated in the wellbore. The rate of salinity buildup was shown to decrease with time as the concentration gradient decreased. Mass buildup rates in the lower well section varied from 12.0 g/min in the first hour to 0.045 g/min during the last five days. The average convective velocity increased to a maximum of 0.23 ft/sec and then decreased below 0.01 ft/sec after 1280 min. The temperature fluctuated 0.5 degree F during the first days of the experiment. The experimental procedure used, and the boundary conditions imposed on the wellbore correspond to the transient phenomena. In order to achieve a steady state concentration gradient the experiment will have to be modified by introducing flow through the aquifer. (MDV)

CONVECTION; DIFFUSION; LABORATORY STUDIES; SALTS; BOREHOLES; TEMPERATURE; CONCENTRATION

<275>
Knapp, R.R., and A.L. Podio, University of Texas, Austin, TX.

Investigation of Salt Transport in Vertical Boreholes and Brine Invasion into Fresh Water Aquifer. 3. Numerical Simulation of Brine Invasion of Fresh Water Aquifers.

CONF-770383; T/OWI/SUB-77/16537; Salt Dissolution, Proceedings of the National Waste Terminal Storage Program Review Meeting, Austin, TX, March 29-30, 1977, (6 pp.), 38 pp. (1977, March)

Evaluation of the applicability of methods for simulating the miscible displacement process using immiscible displacement reservoir simulators was made. The first method using a diffusion coefficient for the fresh water-brine system was found to produce diffusion coefficients that were too large. Therefore the method was considered inapplicable. The next method employed high order accuracy techniques to deal with the diffusion-convection problem. Included in the program are the effects of gravity. Gravitational segregation agrees closely with that exhibited in the experimental studies. Numerical dispersion is partially corrected and, the calculated flood front arrives prematurely. The method also exhibits an oscillation about a complete solvent concentration. These results are encouraging. A similar method should be investigated to determine if it can more effectively reduce the effects of numerical dispersion. (MDV)

CONVECTION; DIFFUSION; COMPUTER PROGRAMS; MODELS; NUMERICAL METHODS; FRESHWATER SYSTEMS; BRINES; AQUIFERS; THEORETICAL STUDIES

<276>
Landstrom, O., K.E. Holmberg, S. Westerberg, and C.E. Klockars, Studsvik Energiteknik AB, Nykoping, Sweden; The Geological Survey of Sweden, Uppsala, Sweden.

In-Situ Experiments on Nuclide Migration in Fractured Crystalline Rocks.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 29-December 1, 1978, (p. 55), 77 pp. (1978)

A field experiment on nuclear migration in fractured rocks was undertaken in Sweden. The tests were conducted in a granite-gneissic center, with radionuclides representing long-lived fission products of the elements selenium, technetium, tin, cesium, iodine, neodymium, and strontium. The mean transit time for the water used was 57 h and the dispersion coefficient was 0.8 sq m/hr. Measurement techniques used were gammaspectrometric borehole logging, continuous activity measurements and water sampling. The tests showed the technetium and iodine travelled as anions with the same velocity as the water, while the strontium ions were retarded by a factor of 6. Cesium and neodymium were not detected, while selenium and tin were completely precipitated in the injection borehole. However, a selenium pulse arrived with a bromine pulse and 20 days later a second selenium breakthrough occurred. With the second

<276> CONT.
selenium breakthrough a tin breakthrough was noted on a shorter flow path. A second series of experiments demonstrated the use of bentonite in the fracture system surrounding the borehole was effective. (NRW)

This paper was presented at the conference but was not published in the Proceedings.

RADIONUCLIDE MIGRATION; RADIONUCLIDES; GRANITES; GNEISSES; LOGGING; BOREHOLES; DISPERSION COEFFICIENT; FIELD STUDIES; BENTONITES; BREAKTHROUGH DISTRIBUTION; RETARDATION FACTOR; METHODS

<277>
Leddicotte, G.W., E.C. Tarnowzer, W.A. Bodger, R.L. Freudberg, and K.W. Horton, Florida Power and Light Company, Miami, FL; Yankee Atomic Electric Company; Nuclear Safety Associates, Bethesda, MD, Utility Waste Management Group.

Suggested Concentration Limits for Shallow Land Burial of Radionuclides.

Waste Management and Fuel Cycles 1978, R.G. Post and K.E. Hacke (Eds.), Proceedings of a Symposium, Tucson, AZ, March 6-8, 1978. Arizona Board of Regents, (pp. 164-174), 641 pp. (1978, March)

This paper is an expansion of one presented at the Atlanta NRC-EPA Conference, May, 1977, which introduced a methodology to develop isotopic limits for the disposal of low-level radioactive wastes via shallow land burial. The basic approach of the earlier effort has been reviewed and reaffirmed. The study has been extended to over 100 isotopes including most, if not all, of practical significance. The effect of daughter products has been included in the analysis. The protection criterion applied to individuals has been made more restrictive. A limit of 5 mrem/yr total body or 15 mrem/yr any organ to a maximum hypothetical individual has been used, instead of the 0.01 MPC in air or water at the point of nearest potential public use adopted in the previous paper. The ten most significant radionuclides, their limiting concentrations, and the limiting pathway for each are as follows: Ra 226 = 0.002 uCi/gm, by erosion into water after 100 yrs (organ), by food grown on site after 100 yrs (whole body); Th 230 = 0.002 uCi/gm, by food grown on site after 10,000 yrs (whole body); Th 232 = 0.004 uCi/gm, by food grown on site after 100 yrs (whole body); I 129 = 0.008 uCi/gm, by food grown on site (organ); C 14 = 0.02 uCi/gm, by food grown on site after 100 yrs (organ); Pa 231 = 0.02 uCi/gm, by food grown on site after 200 years (organ); Th 229 = 0.03 uCi/gm, by food grown on site after 100 yrs (organ); U 238 = 0.03 uCi/gm, by food grown on site after 10,000 yrs (whole body); Tc 99 = 0.06 uCi/gm, by food grown on site after 100 yrs (organ); and Sr 90 = 0.08 uCi/gm, by food grown on site after 100 yrs (organ). The results show that the pathway whereby radioactivity might leave the site by entrance of water into the filled trench to leach activity from the buried waste, and that the leached activity might move with groundwater flow through the subsoil to a point of use (surface stream or well), is the least important of all plausible pathways. This is due to such mechanisms as (1) radioactivity retention within the waste and soil, (2) decay of radioactivity during migration, and (3) watercourse dilution. These mechanisms provide a defense in depth against abrogation of a burial ground's containment of radioactivity. It is concluded that the limiting pathways for all important isotopes are not site dependent, that hydrogeological investigations now underway are not necessary in NRC's or EPA's programs for development of shallow land burial criteria and regulations, that packaging and solidification of reactor wastes can only be justified for those high in Cs 137, Sr 90, or H 3, or for mitigation of transport accidents, and that transuranic limits lower than a few hundred nCi/gm are unnecessary. (Auth; (LKN)

Extensive tabular data is provided in the text. See also G.W. Leddicotte, et al., "Suggested Quantity and Concentration Limits to be Applied to Key Isotopes in Shallow Land Burial," NRC-EPA Conference, Atlanta, GA, May, 1977.

WASTES, LOW-LEVEL, BURIAL, SHALLOW; CONCENTRATION; RADIONUCLIDES; ENVIRONMENTAL EXPOSURE PATHWAY; RELEASE LIMITS; DISPOSAL FACILITIES; INTRUSION; EXCAVATION; INHALATION; INGESTION; EROSION; ACCIDENTS; POLLUTION, WATER; POLLUTION, AIR; SEEPAGE; AGRICULTURE; THEORETICAL STUDIES; PREDICTIONS; RECOMMENDATIONS; GROUND WATER

<278>
Levy, H.B., Lawrence Livermore Laboratory, Livermore, CA.

Methodology for Estimating Accidental Radioactive Releases in Nuclear Waste Management.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 533-539), 563 pp. (1979)

In the assessment of the radiological risks associated with a nuclear waste management system it is necessary to estimate possible accidental radioactive releases so that the consequences of such releases may be evaluated. There are three mechanisms by which radionuclides may be released to the biosphere--air dispersion of fine particles, volatilization, and leaching. One must estimate how much of each significant radionuclide is released by each mechanism. An accidental release of radioactivity generally involves a breach of one or more protective barriers. The probability of breaching protective barriers and what fractions of which radionuclide are released depend on the type of accident occurring and the severity of the accident conditions. The methodology of using these probabilities and fractional releases to derive a set of radioactive release functions for a nuclear waste management system will be discussed in this paper. (Auth)

ACCIDENTS; HAZARD ANALYSIS; WASTE MANAGEMENT; LEACHING; DISPERSION; VOLATILITY; BIOSPHERE; WASTES, RADIOACTIVE

<279>
Logan, S.E., and E.C. Barbano, U.S. Environmental Protection Agency, Office of

<279> CONT.
Radiation Programs, Las Vegas, NV.

Development and Application of a Risk Assessment Method for Radioactive Waste Management. Volume 2. Implementation for Terminal Storage in Reference Repository and Other Applications.

EPA 520/6-78-005; 366 pp. (1978, July)

A Radioactive Waste Management Systems Model, developed and implemented by the University of New Mexico under contract with the U.S. Environmental Protection Agency, is presented. The systems model and associated computer code called ARRAS (Assessment Method for Radioactive Wastes), has two parts. The first part, ARRAS-S, (described in this volume) consists of the Source Term (radioactive inventory versus time), the Release Model, and the Environmental Model. The Release Model considers various geologic and man-caused events which are potential mechanisms for release of radioactive material beyond the immediate environs of a repository or other location; the risk analysis mode uses events distributed probabilistically over time, and the consequence analysis mode uses discrete events occurring at specified times. The site chosen for initial application of ARRAS is the Los Medanos area in southeast NM, under study for potential installation of a Waste Isolation Pilot Plant. For this application, an underground repository area of 10 sq km, providing for accumulation of waste from reprocessing of 187,000 metric tons of spent fuel, is assumed. The most likely geologic event for disturbance of the repository assumed for this assessment appears to be offset faulting, which could interconnect aquifers above and below the bedded salt formation. Annual probability of such faulting is estimated at 1.4 X 10(E-7). This event contributes 9% of the total body dose in Zone 2 (most of Eddy County) and 46% of nonspecific dose, integrated over 700,000 yr. Volcanic explosion and volcanogenic transport to the ground surface are the next most likely release events until 700,000 yr [2.0 X 10(E-12)/yr and 8.1 X 10(E-12)/yr, respectively]. At later times, breakthrough of Rp moves groundwater into the dominant position. The Environmental Model includes: 1) the transport to and accumulations at various receptors in the biosphere, 2) pathways from these environmental concentrations, and 3) resulting radiation dose to man. The highest calculated dose rates to man are obtained for Zone 1 (repository site): 0.006 mrem/yr. The corresponding average nonspecific dose rate (from agricultural products from all zones in the region) is 0.7 man-rem/yr. Application of the model to a repository in shale and to storage of other radioactive and non-radioactive hazardous materials is discussed. (Auth) (LBM)

MODELS; MODELS, MATHEMATICAL; THEORETICAL STUDIES; COMPUTER PROGRAMS; RADIONUCLIDE MIGRATION; NTPP; REPOSITORY; STORAGE, GEOLOGIC; SALT DEPOSITS; SHALES; FAULTS; VOLCANOES; METEORITES; ACCIDENTS; PREDICTIONS; RADIATION DOSE; DOSE RATE; GROUND WATER; ENVIRONMENTAL EXPOSURE PATHWAY; WASTES, RADIOACTIVE; WASTES, NONRADIOACTIVE

<280>
Logsdon, J.E., U.S. Environmental Protection Agency, Office of Radiation Programs, Surveillance and Inspection Division, Washington, DC.

Radioactive Waste Discharges to the Environment from Nuclear Power Facilities.

Radiation Data and Reports 13(3):117-129. (1972, March)

Data relating to discharges of radioactive liquid and gaseous waste have been compiled for 32 selected operating nuclear power facilities. Most experience on radioactive discharge to date has been with pressurized water reactors (PWR) and boiling water reactors (BWR). Limited operating experience has been gained from a high-temperature gas-cooled reactor (HTGR) through operation of Peach Bottom-1. PWRs discussed are: Shippingport, Yankee, Indian Point-1, San Onofre, Connecticut Yankee, and Robert E. Ginna; BWRs discussed are: Dresden-1, Big Rock Point, Humboldt Bay, La Crosse, Nine Mile Point, and Oyster Creek. The data are presented and compared to discharge limits and quantity of electric power produced. In most instances, concentration of radioactivity in waste discharge limits have been maintained at a few percent of the Atomic Energy Commission's licensed discharge limits. Exceptions are mostly associated with either an unusually high percentage of leaky fuel elements or with liquid discharge limits which are artificially low because liquid wastes, in some cases, are not analyzed for radionuclide content. Comparison of power produced to liquid and gaseous waste discharges shows that boiling water reactors discharge relatively large quantities of gaseous waste and pressurized water reactors discharge relatively high quantities of tritium in liquid waste. No obvious trend is discernible concerning the quantity of radioactive waste discharged as a function of power generation. (Auth) (PAP)

WASTES, RADIOACTIVE; WASTES, LIQUID; WASTES, GASEOUS; DISCHARGE; NUCLEAR FACILITIES; REACTORS, BOILING WATER; REACTORS, HIGH-TEMPERATURE GAS-COOLED; REACTORS, PRESSURIZED WATER; TRITIUM; COOLANTS; FISSION PRODUCTS, ACTIVATION PRODUCTS; BETA PARTICLES; RADIATION, GAMMA

<281>
Martin, J.J., and R. Boissonnet, EDF, France.

Radiological Problems of Liquid Waste Disposal into the Sea.

CONF-750411; European Nuclear Society, Proceedings of the 1st Conference, Paris, France, April 21-25, 1975, (pp. 423-428). (1975)

Possible exposure pathways and dosages to man from operation of planned 4000-5000 MW(e) light-water reactors along the French seacoast are being investigated. Many of the parameters used in this study are based on a previous study by the CEA of the spent fuel reprocessing plant at La Hague. A conservative dilution factor for cooling water in the sea is based on a continuous discharge of 160-230 cm s/sec; a study by the Laboratoire National d'Hydraulique of Chatou shows the ratio between sea contamination and cooling water contamination to be greater than 0.1 in an area extending 5 miles above and below the release point along the coast and 1 mile offshore. The population groups to be considered are fishermen, the nearest

<281>

ENVIRONMENTAL TRANSPORT

<281> CONT.

population group, and summer visitors. Relative dose values, not absolute values, are important at this stage. Main doses come from release of Co 60 (60% of total body dose), I 131 (11% of total body dose, 99% of thyroid dose), and Cs 137 (18% of total body dose); the main exposure pathway for total body is extended exposure from sunbathing on shore (sand), and consumption of crustaceans and molluscs for internal exposure. Important concentration factors are 100 Ci/t of I for 1 Ci/cu m of water from consumption of crustaceans and molluscs, 1000 ci/cu m of Cs and I for 1 Ci/cu m of water from sunbathing on sand, and 50,000 Ci/cu m of Co from sunbathing on sand. (LEN)

ENVIRONMENTAL EXPOSURE PATHWAY; BODY COMPARTMENTS; EXPOSURE, EXTERNAL; EXPOSURE, INTERNAL; CONCENTRATION FACTORS; REACTORS, LIGHT WATER; SEAWATER; WASTES, RADIOACTIVE; WASTES, LIQUID; UPTAKE; IODINE; COBALT; CESIUM; FOODS; SEA DISPOSAL; RADIONUCLIDE MIGRATION

<282>

McCarthy, G.J., W.B. White, R. Roy, R.E. Schmitz, S. Komarneni, D.K. Smith, and D.M. Roy, Pennsylvania State University, Materials Research Laboratory, Department of Geosciences, University Park, PA.

Interactions between Nuclear Waste and Surrounding Rock.

Nature 273(5659):216-217. (1978, May 18)

Experimental evidence is presented for the formation of the minerals weeksite and pollucite, by hydrothermal alteration of vitrified radioactive waste and by reaction of calcine with adjacent basalts or shales respectively. A 4 mm spheroid of simulated waste borosilicate glass no. 76-68 (13 wt% fission product oxides) was immersed in water at 300 C and 300 bar in a sealed gold capsule. After 2 weeks, the spheroid had fragmented, and the capsule interior was covered with acicular crystals ($NaFeSi_2O_6$). Also observed were mm-size crystals of Na-Cs-weeksite, $(Na,Cs)_2(UO_2)_2(Si_2O_5)_3.nH_2O$. Approximately 12 wt% of Cs in the glass was taken into solution. In another test, simulated calcine (PW-7a) was placed under 100 bar at 150-200 C in the presence of specimens of shale (Elenora Formation, NV) and of basalt (USGS Standard BCR-1) for 7-30 days. Abundant crystals of Na-pollucite, $(Cs,Na)_2Al_2Si_4O_{12}.nH_2O$, were formed. The latter is a potentially desirable outcome of such interactions, as Cs is pollucite is effectively fixed. Further work is suggested to determine if extraction of U from waste glasses would be observed under reducing groundwater conditions. (LEN)

STORAGE, GEOLOGIC; SILICATES; BASALTS; SHALES; GLASS; HYDROTHERMAL ALTERATIONS; CALCINE; LEACHING; CESIUM; GROUND WATER; WASTES, RADIOACTIVE; WEEKSITE; POLLUCITE; RADIONUCLIDE MIGRATION; REPOSITORY; LABORATORY STUDIES

<283>

Miera, F.R., and T.E. Nakonsca, Los Alamos Scientific Laboratory, Health Division, Los Alamos, NM.

Radiation Doses to Rodents Inhabiting a Radioactive Waste Receiving Area.

Health Physics 34:603-609. (1978, June)

A study was conducted of the gamma ray doses to four species of native rodents inhabiting a low-level radioactive liquid waste disposal area. Absorbed doses of radiation were measured with lithium fluoride thermoluminescent dosimeters that were implanted subcutaneously. The absorbed radiation doses and Cs 137 body burdens were significantly higher for western harvest mice (REITHRODONTOMYS MEGALOTIS) than for deer mice (PEROMYSCUS MANICULATUS), pinon mice (P. TRUEI) and the least chipmunk (EUTAMIAS MINIMUS), reflecting differences in mobility and habitat preferences of the respective species. The average dose received by harvest mice was 26 mrad/day, which was 26% of the highest gamma dose detected at the ground surface in the study plot, although the maximum dose received by individual mice was as high as 45% of the maximum dose rates in the plot. (Auth)

RADIATION, GAMMA; MAMMALS; RODENTS; MICE; CHIPMUNKS; WASTES, LOW-LEVEL; WASTES, LIQUID; WASTE DISPOSAL; ABSORPTION; RADIATION DOSE; DOSIMETERS; CESIUM 137; BODY BURDEN; FIELD STUDIES

<284>

Morizawa, S., T. Inoue, and T. Nakara, Kyoto University, Department of Sanitary Engineering, Kyoto, Japan; Central Research Institute of Electric Power Industry.

Radionuclides Migration in Aerated Zones. 3. Theoretical Estimation of Strontium 90 Migration.

Nihon Genshiryoku Gakkaishi 20(2):133-142. (1978)

A method was proposed for estimating radionuclides migration through an aerated zone with consideration of unsteady soil moisture infiltration process. Two fundamental equations were described, one of which was for soil moisture migration by capillary flow and another one was for radionuclide transportation by soil moisture. This estimation method was examined against experimental results on Sr 90 distribution in a model aerated sand bed as has been reported in detail in previous paper. Some valuable results have been obtained: (1) The proposed method could depict the Sr 90 transportation pattern in a model aerated sand bed. (2) The distribution coefficient kds of Sr 90 between soil and soil moisture depends on soil moisture content. This fact is derived from computer aided numerical experiments. (3) The approximate estimation method proposed by Inoue and Akaki shows large conservative estimation error in Sr 90 migration rate through soil layer. A table summarizing the stated conclusions is included. (Auth) (NDV)

RADIONUCLIDE MIGRATION; STATISTICS; EQUATIONS; MOISTURE; UNSATURATED ZONE; MODELS, MATHEMATICAL; COMPUTERS; DISTRIBUTION COEFFICIENTS; STRONTIUM; THEORETICAL STUDIES; SOILS; GROUND WATER; LABORATORY STUDIES; WASTE DISPOSAL

<285>
Nakamura, H., I. Suzuki, and T. Ueda, National Institute of Radiological Sciences, Chiba, Japan.

Influences of Marine Sediment on the Accumulation of Radionuclides by Green Alga (Ulva pertusa).

Journal of Radiation Research 16(4):228-236.(1975, December)

A study of the distributions of Co 60, Cs 137, Zr 95-Nb 95, and Ru 106-Rh 106 in the green alga ULVA PERTUSA, seawater, and marine sediment was conducted to determine the influence of sediment on the accumulation of radionuclides by the alga. A 3-compartment model was used to analyze the distributions. Influence of the sediment was found to be largest for Co 60, followed by Zr 95-Nb 95, Ru 106-Rh 106, and Cs 137. Activity ratios for Zr 95-Nb 95 and Ru 106-Rh 106 were larger for alga than sediment, while Co 60 and Cs 137 activity ratios were larger for sediment than alga. The Co 60 ratio is approximately 20 times greater for sediment than for the alga. In the case of Cs, most of the activity remained in the seawater. The maximum amount of marine sediment which will reduce the activity accumulated by 0.5 g of green alga was estimated to be 5g/500 ml seawater for Co 60 and more than 10 g/500 ml seawater for the other three radionuclides. Biological half-live based on the transfer coefficients were 10 days for Co 60, 7 days for Cs 137, 26 days for Zr 95-Nb 95, and 28 days for Ru 106-Rh 106. (LKH)

Co 60; Cs 137; Ru 106; Rh 106; Nb 95; Zr 95

COBALT 60; CESIUM 137; SEAWATER; SEDIMENTS; HALF-LIFE, BIOLOGICAL; RADIONUCLIDE MIGRATION; MARINE; DISTRIBUTION; MODELS; UPTAKE; ALGAE; ZIRCONIUM 95; NIOBIUM 95; RUTHENIUM 106; RHODIUM 106; BIOLOGICAL TRANSPORT

<286>
Nelson, J.L., D.H. Denney, and K.C. Knoll, Hanford Atomic Products Operation, Richland, WA.

Hanford Studies in Geochemistry.

TID-7628; Ground Disposal of Radioactive Wastes, J.H. Morgan, Jr., et al. (Eds.), Proceedings of a 2nd Conference, Chalk River, Canada, September 26-29, 1961. U.S. Atomic Energy Commission, Division of Technical Information, Washington, DC, (pp. 218-236), 635 pp. (TID-7628).(1962, March)

The text of this paper concerns the quantitative definition of the variables that affect the retention of radionuclides by soils. Application of chromatographic equations from the literature to soil column breakthrough data permits prediction of breakthrough of radionuclides from ground disposal facilities. A study was made of the effects of temperature, species of accompanying ion, concentration of accompanying ion, concentration of trace ion, pH, column diameter, column length, and flow rate on strontium breakthrough curves. Although initial attempts to fit the column length data to chromatographic equations were not successful, some adjustment methods were found which bring the equations into better agreement with laboratory results. In addition to these findings, this paper reports the results of research on the reactions of rare earths with soils, a study of the effect of hydrodynamic unsaturation on ion exchange, further results from a model crib, and a brief description of the mineral reactions studied at Hanford. (JET)

pH

Sr 90; Co

IONS; SODIUM; CALCIUM; MAGNESIUM; DISPERSION; DIFFUSION; BREAKTHROUGH DISTRIBUTION; ION EXCHANGE; GEOCHEMISTRY; LABORATORY STUDIES; SOILS; pH; CRIBS; MINERALS; EQUATIONS; FIELD STUDIES; ZEOLITES

<287>
Neretnieks, I., Royal Institute of Technology, Department of Chemical Engineering, Stockholm, Sweden.

Analysis of Some Tracer Runs in Granite Rock Using a Fissure Model.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposia, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 819-815), 563 pp.(1979)

While nuclide transport by water in porous media can be described as having mean velocity and dispersion, such a description may not be adequate for fissured rock of low porosity and few and poorly interconnected fissures. A mathematical model based on parallel fissures of varying width with no interconnections was used to analyze some in situ experiments performed. A good fit was obtained for a bromine tracer run, which permitted the evaluation of the mean fissure width distribution and the rock permeability. Laboratory data on adsorption equilibria of strontium on the Studsvik granite were used to predict the retardation of a strontium pulse in the same site. The predicted retardation was in fair agreement with the actual strontium run. The parallel flow model predicts that a pulse will increase in width proportionally to the distance travelled, whereas the normally used statistical dispersion model would predict a pulse which widens proportionally to the square root of the distance travelled. This has a large impact on the time of the first arrival of a radioactive pulse as well as the peak height. (Auth) (SDV)

MODELS, MATHEMATICAL; FISSURES; PERMEABILITY; PREDICTIONS; RADIONUCLIDE MIGRATION

<288>
Nishita, H., H. Hamilton, and L.J. Steen, University of California, Laboratory of Nuclear Medicine and Radiation Biology, Los Angeles, CA.

Extractability of Plutonium 238 and Curium 242 from a Contaminated Soil as a Function of pH and Certain Soil Components: HNO_3-NaOH System.

Soil Science Society of America Journal 42:51-56.(1978)

Aiken clay loam, an acidic, kaolinitic soil, was used to investigate the chemical extractability of Pu 238 and Cm 242 in a

<288> CONT.

HNO3-NaOH system. The soil was treated in a sequential manner for HCl soluble organic matter and salts, decomposed organic matter and dissolved Mn-oxides, free iron oxides, free silica and alumina, amorphous alumino-silicates and residual organic matter. Each treatment was numbered and the tests began with untreated soil. In the untreated soil extractability decreased from pH 1.2 to 2, decreased at slower rate from pH 2 to 4.7, and increased rapidly from pH 7 to 12 after remaining constant between 4.7 and 7. At higher pH extractability decreased. Below pH 3 extractability relative to the control increased. Between 3 and 7 the curve for treatment 2 is very similar to that of treatment 1. The general shape of the curves suggests several stages as the ion passes into the colloidal form. These stages are hydrolysis, precipitation, coprecipitation or adsorption. Increased extractability noted after treatment 3 below pH 6.2 indicated a decrease in the cation exchange capacity from treatment 2 (20.85 to 16.53). For Pu 238 the extractability decreased from pH 1.2 to 4.7, remained relatively constant betw 4.7 and 7.1, increased markedly above 7.1 12.2 and then decreased. In general the extractability curves for Pu 238 were similar to those for Cm 242. The difference between the mean extractabilities of the two radionuclides was significant at the 5% level at all pH levels examined. Distribution coefficients for the unaltered soil ranged from 4.1 x 10(E-3) to 9.1 x 10(E-2) and 1.2 x 10(E-4) to 1.5 x 10 for Pu 238 and Cm 242, respectively. In the acidic pH range CH3COOH-HNOH was more effective than HNO3-NaOH for extraction, and in the alkaline range HNO3-NaOH was more effective. The results indicate that organic matter, free iron oxides and free silica, alumina, and amorphous alumino-silicates influenced the chemical extractability of Pu 238 and Cm 242. (NDV)

pH; Ion Exchange Capacity; Distribution Coefficient

Pu 238; Cm 242

ADSORPTION; CATION EXCHANGE CAPACITY; CHELATES; CHEMICAL PROPERTIES; COLLOIDS; COMPLEXES; DISTRIBUTION COEFFICIENT; KAOLINITE; LABORATORY STUDIES; ORGANIC COMPOUNDS; pH; PLUTONIUM; PRECIPITATION, CHEMICAL; SILICATES; ALUMINUM; SOILS; CLAYS; LOAM; WASTES, RADIOACTIVE; CURIUM; HYDROLYSIS; EXTRACTION

<289>

Onishi, Y., E.M. Arnold, R.J. Serne, C.E. Cowan, F.L. Thompson, and D.W. Mayer, Battelle-Pacific Northwest Laboratories, Richland, WA.

Annual Report October 1977 to September 1978 Mathematical Simulation of Sediment and Contaminant Transport in Surface Waters.

NUREG/CR-0656; PNL-2902; 32 pp. (1979, January)

Both published and unpublished information was critically reviewed to evaluate the state-of-the-art of mathematical modeling of radionuclide transport in surface waters, and current knowledge on adsorption and desorption mechanisms of radionuclides with sediments in surface waters. Based on the literature survey, a mathematical model capable of predicting migration and accumulation of radionuclides in marine environments was synthesized from available models. The PETRA code was chosen because it is applicable to both dissolved and particulate radionuclide transport. PETRA was originally developed for rivers and then extended to estuaries. The model was modified to make it applicable to oceans and the great lakes by including wave motion to suspended marine sediments. The modified PETRA code is an unsteady (as well as steady), two-dimensional finite model, utilizing the Galerkin weighted residual method. The model consists of three submodels coupled to include the effects of sediment contaminant interaction on contaminant transport. The submodels are: 1) sediment transport model, 2) dissolved contaminant transport model, and 3) particulate contaminant (contaminant adsorbed by sediment) transport model. (RAF)

MODELS, MATHEMATICAL; SEDIMENTS; RADIONUCLIDE MIGRATION; SURFACE WATERS; SORPTION; OCEANS; EQUATIONS; MARINE SYSTEMS; LAKES; RIVERS; ESTUARIES; PARTICLES; CONVECTION; DISPERSION; RESUSPENSION; WAVES; DISTRIBUTION COEFFICIENT

<290>

Price, S.M., and L.L. Ames, Atlantic Richfield Hanford Company, Physical and Life Science Technology Section Research Department, Richland, WA; Battelle-Pacific Northwest Laboratories, Richland, WA.

Characterization of Actinide-Bearing Sediments Underlying Liquid Waste Disposal Facilities at Hanford.

ARH-SA-232; IAEA-SM-199/87; CONF-751105; Transuranium Nuclides in the Environment, Proceedings of a Symposium, San Francisco, CA, November 17-21, 1975. (1975, September)

Past liquid waste disposal practices at the U.S. Energy Research and Development Administration's Hanford Reservation have included the discharges of solutions containing trace quantities of actinides directly into the ground via structures collectively termed "trenches". Characterization of samples from two of these trenches, the 216-Z-9 and the 216-Z-1A(a), has been initiated to determine the present form and migration potential of plutonium stored in sediments which received high salt, acidic waste liquids. Analysis of samples acquired by drilling has revealed that the greatest measured concentration of Pu, about 10(E+6) Pu 239/liter of sediment, occurs in both facilities just below the points of release of the waste liquids. This concentration decreases to about 10(E+3) uCi Pu 239/liter of sediment within the first 2 meters of the underlying sediment column and to about 10 uCi Pu 239/liter of sediments at the maximum depth sampled (9 meters). Examination of relatively undisturbed sediment cores illustrated two types of Pu occurrence responsible for this distribution. One of these types is composed of Pu particles (>70 wt% PuO2) added to the disposal site in the same form. This "particulate" type was "filtered out" within the upper 1 meter of the sediment column, accounting for the high concentration of Pu/liter of sediment in this region. The second type of Pu (<0.5 wt% PuO2) was originally disposed of as soluble Pu(IV). This "nonparticulate" type penetrated deeper within the sediment profile and was deposited in association with silicate hydrolysis of

<290> CONT.
the sediment fragments. (Auth)

ACTINIDES; SEDIMENTS; WASTES, LIQUID; NUCLEAR FACILITIES; WASTE DISPOSAL; TRENCHES; RADIONUCLIDE MIGRATION; PLUTONIUM 239; AMERICIUM 241; OXIDES; PARTICLES; VALENCE; SAMPLING; DISPOSAL SITE; FIELD STUDIES

<291>
Put, M., and H. Bernaens, Battelle-Pacific Northwest Laboratories, Richland, WA.

Mathematical Modelling of Radionuclide Migration in a Homogeneous Clayey Formation (Application to the Case of the Boom Clay at Mol).

CONF-770565; BNWL-tr-273; Risk Analysis and Geologic Modelling, Proceedings of a Workshop, Ispra, Italy, May 23-27, 1977, (pp. 116-129). (1977, October 28)

A simple three-space model was developed by CEN from evaluation of migration of actinides and fission products in the ground. Ground measurements and laboratory analyses for the Boom clay were used to check the model. The Boom clay is an Oligocene homogeneous clayey formation, that lies between the depths of -160 and -270 m at CEN. It is underlain by an aquiferous sandy formation with low permeability, and is overlain by Mio-Pliocene sands which are aquifers. Experimental data indicate that the hydraulic gradient is 2 m of water column and any vertical circulation stems from the Mio-Pliocene sands through the clay to the Oligocene sands. The porosity of the clay is 22.7%, density 1.93 ton/cu m, and permeability coefficient 3.6 x 10(E-3) m/yr. Distribution coefficients range from 100 to 3370. Partial differential equations are used in the model to describe the concentration of radionuclides along the three axes with a starting rectangular pulse. Radioactive decay and delay time resulting from the containers are taken into account. The model may be used to determine when the maximum concentration of radioactive ion is equal to the maximum admissible concentration. Basic assumptions for this evaluation were: an activity of about 70 Ci/cu m, containers with the dimension of 30 cm diameter and 1.5 m height; dispersion coefficient in the x direction is 5 x 10(E-6) sq cm/yr and in the y and z directions values 100 times smaller; and the maximum permissible concentration of Pu 239 is 5 x 10(E-6) Ci/cu m. From calculations it would take Pu 239 250,000 years to move 18 m in the Boom clay. The model will be adapted to particular cases of geological nonhomogeneity. (BDV)

Stratigraphic Unit Thickness; Hydraulic Gradient; Porosity; Density; Permeability Coefficient; Distribution Coefficient

ANISOTROPY; AQUIFERS, CONFINED; CLAYS; COMPUTER CODES; CONCENTRATION; CONTAINERS; DENSITY; DEPTH; DISPERSIVITY; DISTRIBUTION COEFFICIENT; EQUATIONS; FIELD STUDIES; FISSION PRODUCTS; GEOLOGY; GROUND WATER; MAXIMUM PERMISSIBLE CONCENTRATION; MODELS, MATHEMATICAL; RADIONUCLIDE MIGRATION; SANDS; WASTE DISPOSAL; WASTES, TRANSURANIC; WASTES, SOLID; LABORATORY STUDIES

<292>
Duncan, D., and L. Appleby (Translator), Battelle-Pacific Northwest Laboratories, Richland, WA.

Contamination of Soils: Behavior of Radioactive Ions in Unsaturated Soils after Water Disposal into a Limited Surface, Part 2.

(1973, February)

A study of the contamination of unsaturated soils as related to the wetness of the zone is presented. Iodine 131 and Sr 85 were used to typify the retention of a loosely held radionuclide and of a strongly fixed one. For I 131 the movement of the water and the nuclide are coincident, and fixation occurs during dryout. A strong concentration of I 131 is produced at the surface with an important activity gradient between 0 and 30 cm. No decontamination of the surface occurs after intense elution. The Sr 85 diffuses slowly toward the bottom of the soil and tends to accumulate at the surface. When washed with large quantities of water only a small fraction of Sr 85 was eluted. However when the water is charged with Ca more Sr 85 is entrained. The experiments show that when there is no liquid transfer by infiltration or redistribution the contaminants are confined. The fluid-soil interface and the radionuclide-soil interface can be measured with the latter always behind the former. Furthermore, the importance of ion presence and concentration are demonstrated in this series of experiments. (BDV)

I 131; Sr 85; H 3

CONTAINMENT; CONTAMINATION; DIFFUSION; EQUATIONS; FIXATION; INFILTRATION; IONS; RADIONUCLIDE MIGRATION; SOILS; WASTES, LIQUID; WASTES, RADIOACTIVE; STRONTIUM; CESIUM; UNSATURATED ZONE; LABORATORY STUDIES

<293>
Routson, R.C., G. Janzen, and A.V. Robinson, Battelle-Pacific Northwest Laboratories, Richland, WA.

Americium 241, Neptunium 237, and Technetium 99 Sorption on Two United States Subsoils from Differing Weathering Intensity Areas.

Health Physics 33(4):311-317. (1977, September)

For the purpose of assessing the consequences of a potential release event from a high-level waste repository, the sorption of Am 241, Np 237, and Tc 99 was measured on two soils as a function of solution concentration of Na and Ca from extremes of weathering intensity areas. Distribution coefficients (Kd values) were determined on subsoil samples from Washington and South Carolina for Am 241, Np 237, and Tc 99, as a function of equilibrium solution concentration of calcium Ca(+2) and of sodium Na(+). Kd values decreased in all cases with increasing solution concentrations of Ca(+2) and Na(+). For the South Carolina subsoil, Kd values ranged from 1.0 to 67 ml/g for Am 241 as a function of Ca(+2), from 0.2 to 0.002 M, respectively; 1.6 to 28 ml/g for Am 241 as a function of Na(+); 0.43 to 0.66 ml/g for Np 237 as a function of Ca(+2); and 0.16 to 0.25 ml/g for Np 237 as a function of Na(+), from 3.0 to 0.015 M, respectively. For the Washington soil, Kd values were > 1200 ml/g for Am 241 and ranged from 0.36 to 2.37 ml/g

<293>

ENVIRONMENTAL TRANSPORT

<293> CONT.
as a function of Ca(+2) and from 3.10 to 3.90 ml/g for Np 237 as a function of Na(+), over the above concentration ranges. Kd values for Tc 99 were essentially zero at all NaHCO3 concentrations on the South Carolina subsoil. This study demonstrated that sorption assessments based on plutonium data are inadequate for estimating other actinide elements sorption. Am 241 and Np 237 were found to be poorly sorbed in comparison to Pu. (Auth)

Distribution Coefficient

Am 241; Np 237; Tc 99

AMERICIUM 241; NEPTUNIUM 237; TECHNETIUM 99; SORPTION; SOILS; SANDS; WEATHERING; DISTRIBUTION COEFFICIENT; CALCIUM; SODIUM; PLUTONIUM; CATION CONCENTRATION; ION EXCHANGE CAPACITY; LABORATORY STUDIES

<294>
Bostson, R.C., G. Jansen, and A.V. Robinson, Battelle-Pacific Northwest Laboratories, Richland, WA.

Sorption of Technetium 99, Neptunium 237, and Americium 241 on Two Subsoils from Differing Weathering Intensity Areas.

BNWL-1989; 15 pp.(1975, May)

A coarse textured, sandy, low exchange capacity subsoil from a semi-arid portion of eastern Washington and a moderate exchange capacity, humid-environment subsoil from South Carolina were compared in terms of their sorption of selected radionuclides. The Washington soil contained 0.8 mg/g CaCO3, was 10.5% silt and 0.5% clay, and had a pH of 7.0. The South Carolina soil h . less than 0.2 mg/g CaCO3, 3.6% silt, 37.2% clay, and a pH of 5.1. Sorption of Am 241, Np 237, and Tc 99 was measured in a batch equilibrium system using solutions of calcium nitrate, sodium nitrate, and sodium bicarbonate (for Tc). Measurements of Am and Np were made from planchettes in a 2 pi gas flow system; Tc was measured by liquid scintillation counting. Distribution coefficients from the Washington subsoil were greater than 1200 for Am 241 as a function of Ca++ and Na+, 0.36-2.37 for Np 237 as a function of Ca++, and 3.19-3.90 for Np 237 as a function of Na+. South Carolina values were 1.0-67 for Am 241 as a function of Ca++, and 0.16-0.25 for Np 237 as a function of Na+. Tc 99 distribution coefficients were essentially zero at all NaHCO3 concentrations in South Carolina subsoil. (LKH)

Distribution Coefficient; Calcium Carbonate Content; Clay Content; pH

Am 241; Np 237; Tc 99

SORPTION; WASTES, HIGH-LEVEL; WASTES, TRANSURANIC; CALCIUM; SODIUM; LABORATORY STUDIES; AMERICIUM; NEPTUNIUM; SOILS; TECHNETIUM; EQUATIONS

<295>
Savannah River Laboratory, Aiken, SC.

Retention of Organic-Complexed Plutonium in Soil.

DPST-75-125-3; Savannah River Laboratory Quarterly Report, Waste Management, July-September, 1975, (pp. 27-29), 40 pp.(1975)

A laboratory study was undertaken to determine the mechanism of migration of the solvent-complexed plutonium through the soil. The plutonium contained in retired separations process solvent, is stored in underground tanks. About 175,000 gallons of waste containing 92 Ci plutonium is stored. Tests show that the average groundwater/solvent distribution coefficient is 0.005. An equilibrium equation was used to evaluate a postulated large release of organic-complexed plutonium to the subsurface soil. Results indicate that the soil would sorb Pu and the isotope would move very slowly through the soil-water system with a sorption distribution coefficient of about 150. It would require 50,000 years for the Pu to reach Four Mile Creek and by that time all of the Pu 238 would have decayed. Most of the Pu would decay within the Savannah River plant boundaries. No further study is planned. (EDV)

Waste Volume; Distribution Coefficient

Pu 238; Pu 239

SORPTION; PLUTONIUM; CONTAINMENT; CONTAMINATION; DISPOSAL SITE; DISTRIBUTION COEFFICIENT; GROUND WATER; LABORATORY STUDIES; EQUATIONS; SOILS; TANKS; WASTE STORAGE; WASTES, LIQUID

<296>
Science Applications, Inc., Oak Ridge, TN.

Technical Support for Generic Environmental Impact Statement: Radioactive Waste Isolation in Geologic Formations. Volume 23 - Environmental Effluent Analyses.

Y/OWI/TM-36-23; 125 pp.(1978, April)

This report analyzes sources and magnitudes of releases of radioactive and non-radioactive effluents to the environment as a result of construction of a radioactive waste repository and subsequent waste handling operations. Solid effluents arise mainly during construction of the underground facility in which large quantities of excavated rock will be brought to the surface. Other sources of solid effluents would be surface excavation during construction, and refuse and sanitary wastes during operation. All solid effluents are expected to be non-radioactive except the worn out equipment and accessories from contaminated waste-handling areas. Liquid effluents would arise during construction from surface runoff and sewage wastes. These liquid effluents are not expected to be an environmental concern. During operation, however, decontamination procedures and, in the case of non-salt rocks, groundwater inflow could give rise to contaminated liquid effluents. Decontamination liquids and ground water, if contaminated, could require treatment to comply with EPA standards before being discharged. Airborne effluents result from mining activities, surface construction, and waste-handling operations. Construction related airborne effluents would be vehicular exhausts, naturally occurring gases from the mine, dust, and emissions from coal-fired steam generators. Except for naturally

<296> CONT.
occurring Rn gas, these effluents are non-radioactive. Airborne radioactive effluents could result from normal waste handling and storage. Small amounts of container surface contamination could be released to the facility ventilation system and to the facility stack, but even by conservative assumptions there should be no appreciable dose at the site fenceline. It was determined that radiolytic hydrogen production during waste storage poses no potential fire hazard in the current repository design. Accidents to waste containers were considered and appropriate scenarios defined; the resulting airborne releases were assessed using conservative assumptions. The worst case involved concentrations at the site boundary of about 700 times the NRC limit for unrestricted areas considering OWI-defined waste, and about 100 times the limit considering BPNL-defined waste. Hydrogen generation was again investigated, using intermediate-level BPNL wet waste fixed in bituminous material. It was concluded that the worst situation would result in concentrations of hydrogen in air of about 3 orders of magnitude below the combustible limit. (Auth)(LKH)

REPOSITORY; EXCAVATION; EFFLUENTS, AIRBORNE; WASTES, NONRADIOACTIVE; WASTES, ORGANIC; WASTES, SOLID; WASTES, LIQUID; EFFLUENTS, LIQUID; DECONTAMINATION; GROUND WATER; CONSTRUCTION; DUSTS; GASES; SURFACE CONTAMINATION; CONTAINERS; WASTE PROCESSING; RADIOLYSIS; ACCIDENTS; FIRE HAZARDS; THEORETICAL STUDIES; STORAGE, GEOLOGIC

<297>
Seitz, M.G., and R.A. Couture, Argonne National Laboratory, Chemical Engineering Division, Argonne, IL.

Trace-Element Transport in Lithic Material by Fluid at High Temperatures.

ANL-78-68; Chemical Engineering Division Fuel Cycle Programs Progress Report, January-March 1979, M.J. Steindler, et al., Section 8, (pp. 183-191), 242 pp.(1979, April)

To study the migration of trace elements in natural geologic formations, equipment has been assembled for pumping solutions through packed mineral columns at elevated temperatures. Two columns, each 5 cm long and 1 cm in diameter, were packed with kaolinite. A slurry of kaolinite was pumped into the columns, which were each fitted with a stainless steel (type 316) frit at the bottom. The porosity of column 1 was 44%; porosity of column 2 was 49%. In the first attempt to trace the movement of a solution through a column, it was found that the chloride in a potassium chloride spike was not eluted - even after more than 15 free-column volumes of flow. The chloride was assumed to have been bound in the column. Experiments were conducted to determine if iodine may be similarly bound. In the first experiment, which was run at room temperature and at a flow rate of 0.025 ml/min, about 10% of the iodine was eluted without interacting with the column. The remainder interacted and was retarded in relation to the water front. Three major peaks and 1-3 smaller peaks were detected, suggesting that the iodine was present as several chemical species which did not exchange appreciably during their contact with the kaolinite (about 1 hr). The experiment was duplicated in a second column, yielding results in good agreement with those of the previous experiment (about 20% of the iodine was eluted in the first peak). The results of the second experiment suggest that only two slowly exchanging forms of iodine were present, rather than several. Iodine was eluted through the second column again, this time at 132 C and a flow rate of 0.10 ml/min. Only one peak was found, and the iodine eluted quickly and completely. A spike of Cs 134 was injected into a stream of 0.001 M NaHCO3, which was pumped through a kaolinite column at 0.0250 ml/min. No Cs was found in the eluate after 89 free column volumes, so the clay was extruded from the column, sliced, and analyzed for Cs activity. Most of the Cs was bound to the top 0.13 g of clay. No Cs migrated through more than 2 g of clay; thus, the bottom 63% of the column contained no Cs. The data indicate that kaolinite acts as a barrier to iodine migration in near-neutral solutions. However, because of the complex behavior, the retardation of iodine cannot be represented by a single partitioning (Kd) parameter. Nor is it certain that the migration of iodine can be represented by a single partitioning parameter for each of the chemical species of iodine because exchange between the species at rates yet to be determined may significantly modify the shapes of the curves. Also, the migration properties of iodine are sensitive to temperature, and at 132 C, all retardation of iodine by the clay is lost. Even at temperatures slightly above ambient, the migration of iodine apparently is not the same as a rock temperature. These results stress the importance of determining migration properties under dynamic conditions and at various temperatures. (LKH)

Cs 134; I 131

RADIONUCLIDE MIGRATION; ROCKS; SEDIMENTS; LABORATORY STUDIES; THERMAL PROPERTIES; KAOLINITE; CLAYS; IODINE 131; CESIUM 134; SORPTION; PARTITIONING; EQUIPMENT

<298>
Seitz, M.G., P.G. Rickert, S. Fried, A.M. Friedman, and M.J. Steindler, Argonne National Laboratory, Chemical Engineering Division and Chemistry Division, Argonne, IL.

Transport Properties of Nuclear Waste in Geologic Media.

ANL-78-68; Chemical Engineering Division Fuel Cycle Programs Progress Report, January-March 1979, M.J. Steindler, et al., Section 7, (pp. 176-182), 242 pp.(1979, April)

Infiltration experiments are described which use a new apparatus capable of delivering a uniform and controlled flow of fluid at high pressures to a column of rock material. A system of strontium in water and glauconite was studied, because Sr is expected to exhibit simple migration behavior in glauconite. The hydrodynamic properties of a mineral column made from the sand were measured by eluting tritiated water through the column. Elution of a 20 ml sample of tritiated water through the glauconite column yielded a sharp, symmetrical curve with the peak at the eluate volume corresponding to the calculated free volume of the apparatus. This curve indicates that there is little dispersion in the eluate produced by mixing in the fluid stream. Exchange properties of

<299>

ENVIRONMENTAL TRANSPORT

<298> CONT.
the column were investigated by monitoring Sr 85 migration through the column. Elution at flow rates up to 1 ml/min yielded symmetrical curves with widths considerably greater than that for tritium. The results indicate that Sr reacts with the mineral column and rapidly exchanges between the solution and solid mineral. The reaction of Sr results in characteristics of migration that are predicted by simple chromatographic theory - that is, the Sr moves in a peak with a well-defined velocity. The velocity of Sr estimated from the elution curve is 0.05 times the velocity of the water front or 45 m/yr in glauconite. (LKN)

Migration Rate; Flow Rate

Sr 85; H 3

GLAUCONITE; STRONTIUM 85; TRITIUM; LABORATORY STUDIES; RADIONUCLIDE MIGRATION

<299>
Seitz, M.G., P.G. Rickert, S.N. Fried, A.N. Friedman, and M.J. Steindler, Argonne National Laboratory, Chemical Engineering Division, Argonne, IL; Argonne National Laboratory, Chemistry Division, Argonne, IL.

Studies of Nuclear-Waste Migration in Geologic Media: Annual Report, November 1976-October 1977.

ANL-78-8; 81 pp. (1978, March)

The confinement of nuclear wastes in geologic formations is being considered as a method of permanently disposing of the waste. Laboratory experiments (column infiltration, static absorption, and batch partitioning experiments) were performed with nuclides of cesium, plutonium, neptunium, and americium. The purpose of the experiments was to determine the migratory characteristics of long-lived radionuclides that could be mobilized by groundwaters infiltrating a nuclear waste repository and the surrounding body. In the column infiltration experiments, the positions of peak concentration of cesium in chalk or shale columns; plutonium in a limestone; americium in limestone, sandstone, or tuff; and neptunium in a limestone column did not move when the columns were infiltrated with water. However, fractions of each of the nuclides were observed downstream from the peaks, indicating a large dispersion in the relative migration rates of each of the trace elements in the rock materials studied. The results of the static absorption experiments indicated that plutonium and americium are strongly absorbed from solution by the common rocks studied and that their migration relative to ground water flow is thereby retarded, a conclusion consistent with results of the column infiltration experiments. Batch partitioning experiments were performed to test whether absorption processes were reversible. After granulated basalt and americium-bearing water were contacted in an absorption step, part of the water was replaced with water free of americium and the americium repartitioned between rock and solution. The distribution of americium after desorption was comparable to its distribution after absorption. In contrast, when tablets of various rocks were allowed to dry between adsorption and desorption tests, plutonium and americium were not generally desorbed from the tablets. This suggests that reversible reactions of nuclides between waters and rocks may be upset by treatments such as drying. In batch partitioning experiments with plutonium- and americium-bearing water and granulated basalt of several different particle sizes, the partitioning of americium and plutonium did not correlate with the calculated area of the fracture surfaces nor did the partitioning remain constant. Partitioning is postulated to be a bulk phenomenon with complete penetration of the 500 micrometer size and smaller particles. (Auth) (JWT)

LABORATORY STUDIES; AMERICIUM; PLUTONIUM; NEPTUNIUM; CESIUM; BASALTS; INFILTRATION; ABSORPTION; DESORPTION; GEOLOGY; ROCKS; LIMESTONES; SHALES; SANDSTONES; TUFFS; WASTES, RADIOACTIVE; PARTITIONING; REPOSITORY; STORAGE, GEOLOGIC

<300>
Serne, R.J., D. Rai, R.J. Mason, and M.R. Molecke, Battelle-Pacific Northwest Laboratories, Richland, WA.

Batch Kd Measurements of Nuclides to Estimate Migration Potential at the Proposed Waste Isolation Pilot Plant in New Mexico.

PNL-2848; 81 pp. (1977, October)

Seven samples from cores of strata occuring in the subsurface at the WIPP site in southeast NM were studied to determine the sorption distribution coefficients (Kd) of Cs, Sr, Tc, Ru, Sb, Ce, Eu, Pu, Np, Cm, Am, U, and Pa in rocks potentially in contact with radioactive wastes. The samples consisted of dolomites of the Salado Formation (Celeba-853' and Magenta-738'), limestone (#1 Cherry Canyon), arkosic sandstones (#2 and #3 Cherry Canyon), and halites (Salt Core-2726' and Salt Core-2750'). Measurements were made by the batch method on crushed material in brine, groundwater, and distilled water solutions. Results show that sorption of Sr, Tc, Ru, Sb, Cs, and U was poor (less than 5 ml/g) for all lithologies in brine solutions; sorption of Eu, Ce, Pa, Np, Pu, Am, and Cm is considerably higher (greater than 1000 ml/g for Am and Cm). Sorption values in groundwater (ionic salt concentration 25 x the equilibrated distilled water) were low for Sr and Cs. Kd's for rocks in distilled water ranged from 20-1000 ml/g for Sr, Ru, Cs, Ce, and Eu; no values were measured for Pu, Np, Am, and Pa in distilled water. Recommendations are that more work be performed with actinides in geologically realistic solutions with ionic strengths intermediate to the distilled water and brine concentrations, and that water velocities, waste leach rates, time periods to reach the biosphere, and radionuclide transport and retardation mechanisms be determined in order to assess the hazard to man. (LKN)

Photomicrographs of rock sample thin sections and information on the composition of the carrier solutions are given in appendices.

Distribution Coefficient

Pa 233; Np 237; Eu 152; Ce 144; Sb 125; Sr 85; Cs 137; Ru 106; Tc 95m; Am 241; Pu 237; Cm 243; U 235

CORES; SORPTION; ACTINIDES; SEDIMENTARY ROCKS;

<300> CONT.
SALT DEPOSITS; RADIONUCLIDES; DOLOMITE; SANDSTONES; LIMESTONES; GROUND WATER; DISPOSAL SITE; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; RADIONUCLIDE MIGRATION; WIPP; STORAGE, GEOLOGIC; BRINES; AQUIFERS; ION EXCHANGE; ADSORPTION; WASTE MANAGEMENT; WASTE DISPOSAL; WASTE STORAGE; MINERALS; LABORATORY STUDIES

<301>
Sheppard, J.C., J.A. Kittrick, M.J. Campbell, and T.L. Hardt, Washington State University, Pullman, WA.

Determination of Distribution Ratios and Diffusion Coefficients of Neptunium, Americium, and Curium in Soil-Aquatic Environments.

RLO-2221-T-12-3; 17 pp. (1977, August)

Research was completed on the determination of distribution ratios of Am(+3), Cm(+3), and Np(+5) between 12 U.S. soils and water. Average distribution ratios for all soils were 3.18 + or - 0.66 for Am 241, 3.32 + or - 0.70 for Cm 244, and 1.87 + or - 0.45 for Np 237. Since much of the observed distribution ratio was due to suspended soil particles, a new effective distribution ratio was defined to account for these suspended particles. It was found that the effective distribution ratio was very dependent on centrifuging and sampling conditions and explained why it is difficult to obtain agreement between experiments and experimenters. A separation function was derived which takes these factors into consideration. In addition, an equation was developed which explains the effective distribution ratio in terms of the separation factor, the particle size distribution of the soil, and the extent or nature of the binding of actinide elements to soil particles. The diffusion coefficients of Am(+3) and Np(+5) in 12 moist U.S. soils were determined; they were: 11.1 X 10(E-11)-1.18 X 10(E-11) for Am 241 (+3) and 9.03 X 10(E-8)-1.49 X 10(E-9) sq cm/sec for Np 237 (+5). Equations were developed to calculate Cm(+3) diffusion coefficients from x-ray counting data. Diffusion coefficients for Cm(+3) were determined for six U.S. soils [6.2 X 10(E-11)-0.6 X 10(E-11) sq cm/sec]. Within the uncertainty of the experimental counting data, x-ray absorption coefficients, the composition of the soil-water mixture, and the possibility of changes in the chemical form of the Cm(+3) species, the Cm(+3) diffusion coefficients agree with the comparable Am(+3) data. This is expected because the hydrated radii of Am(+3) and Cm(+3) are very similar. The major finding of this year's research was that the distributions of Am, Cm, and presumably Pu cations between the aqueous and soil phases indicate that small (less than 1000 um radius) easily transported, actinide-bearing particles are responsible for most, if not all, of their distribution, as opposed to free ions as previously believed. (Auth) (LKM)

Grain Size Distribution; Diffusion Coefficient

Am 241; Cm 244; Np 237

EQUATIONS; AMERICIUM; NEPTUNIUM; CURIUM; SOILS; PARTICLES; RADIONUCLIDE MIGRATION; ADSORPTION; LABORATORY STUDIES; WATER; GROUND WATER; SOIL TRANSPORT; ACTINIDES; COLLOIDS; DISPERSION COEFFICIENT; DIFFUSION

<302>
Staley, G.E., G.P. Turi, and D.L. Schrieber, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, DC.

Radionuclide Migration from Low-Level Waste: A Generic Overview.

(1977)

The objective of this study was to evaluate generically the sensitivity of radionuclide migration to certain transport parameters, as related to shallow low-level radioactive waste burial sites. There are many site-specific studies of radionuclide migration through groundwater reported in the literature; however, no quantitative generic study of the effects of transport parameters could be found. This study could serve as a conservative numerical basis for establishing site boundaries. A range of soil types that could exist as such waste burial sites was considered in the analyses. A realistic value of hydraulic conductivity, total porosity, effective porosity, and bulk density was selected for each soil type. Soils were assumed to be completely saturated. For each soil type, the time-varying concentrations of specific radionuclides were determined at given distances from the source. Sensitivity analyses were performed for each soil type by varying the hydraulic gradient of the water table and the leach rate of the source. The sensitivity of distribution coefficients to pH was investigated for only the silt soil and three radionuclides, because of limited available data. Conservative analyses were conducted by considering only time decay of radionuclides resulting from ground water travel and holdup due to ion-exchange with soil particles. Realistic analyses were performed using a three-dimensional dispersion model. The reported analyses assimilate the available field data into a practical generic application, i.e., a first step in establishing hydrologic siting criteria for low-level radioactive waste facilities. (Auth)

Hydraulic Conductivity; Hydraulic Gradient; Porosity; Density; Distribution Coefficient; Leaching Rate; pH

Cr 51; Co 202; Mn 54; Cs 134; Fe 55; Co 60; H 3; Pu 241; Cm 244; Sr 90; Cs 137; Pu 238; Am 241; Pu 239; Ni 59; Tc 99; Np 237; I 129

RADIONUCLIDE MIGRATION; WASTES, LOW-LEVEL; BURIAL; GROUND WATER; SOILS; SILTS; CLAYS; SANDS; PARTICLE SIZE; LOESS; TILL; SITE SELECTION; HYDRAULIC CONDUCTIVITY; POROSITY; DENSITY; RADIONUCLIDES; HYDRAULIC GRADIENT; WATER TABLE; LEACHING; DISTRIBUTION COEFFICIENT; pH; ION EXCHANGE; MODELS; DISPERSION; SORPTION; TIME FACTOR; EQUATIONS; AQUIFERS; TRITIUM; STRONTIUM; CESIUM; PLUTONIUM; RETARDATION FACTOR

<303>
Sugiura, Y., K. Saruhashi, and Y. Miyake, Meteorological Research Institute, Tokyo, Japan; Geochemistry Research Association, Tokyo, Japan.

Evaluation on the Disposal of Radioactive Wastes into the North Pacific.

Papers in Meteorology and Geophysics

<303>

ENVIRONMENTAL TRANSPORT

<303> CONT.
27(3):81-87.(1976, September)

Evaluation of deep sea disposal of radioactive waste into the North Pacific on the basis of diffusion processes is done using a simplified model. The results of calculation by employing different sets of diffusion coefficients on the horizontal and vertical directions show that for the smaller diffusion coefficients discrepancies between relative concentration of nuclides and the law of radioactive disintegration are greater. For larger coefficients, concentration above the site of release approaches mean, homogeneous concentration which satisfies the rule of radioactivity. From oceanographic and radiological view points, it is concluded that 10 sq cm/sec and 10(E+8) sq cm/sec respectively of diffusion coefficient of the vertical and horizontal directions are appropriate and reasonable in the North Pacific for the purpose of dose evaluation. From this a new definition of high-level waste unsuitable for sea disposal was obtained assuming a daily seafood intake of 100 g/day, and concentration factors in seafoods of 80 for Co 60, 1 for H 3, 0.2 for Sr 90, 1 for Ra 226, and 10 for Pu 239. Maximum permissible annual rates of radionuclide release into the North Pacific (Ci/yr) are 8 X 10(E+6) for Co 60, 2 X 10(E+10) for H 3, 2 X 10(E+6) for Sr 90, 2 X 10(E+6) for Cs 137, 2 X 10(E+2) for Ra 226, and 6 X 10(E+2) for Pu 239. Assuming a maximum release of solid radioactive wastes into the North Pacific of 10(E+6) tons/yr, acceptable limits are as follows: no more than 10 Ci/ton for beta-gamma emitters (except tritium), 1 Ci/ton for Sr 90 plus Cs 137, and 10 (E+4) Ci/ton for tritium; no long-lived alpha emitters may be dumped. These values are about 100 times less than those set by the IAEA. (Auth) (LKH)

MODELS; MODELS, MATHEMATICAL; SEA DISPOSAL; WASTES, HIGH-LEVEL; DIFFUSION; OCEANOGRAPHY; FOODS; RELEASE LIMITS; COBALT; TRITIUM; STRONTIUM; RADIUM; PLUTONIUM; CESIUM; THEORETICAL STUDIES; RADIONUCLIDE MIGRATION; WASTES, RADIOACTIVE; EQUATIONS; SITE SELECTION; DISPERSION COEFFICIENT

<304>
Talbert, D.M., and D.R. Anderson, Sandia Laboratories, Albuquerque, NM.

United States Seabed High-Level Radioactive Waste Disposal Feasibility Study.

Transactions of the American Nuclear Society 26:297-298; CONF-771109; American Nuclear Society Annual Meeting, New York, NY, June 12-17, 1977, (pp. 297-298).(1977)

Sandia Laboratories is conducting a multidisciplinary investigation of the feasibility of high-level waste disposal in submarine geologic formations. Early program efforts were directed at assembling existing data. The second phase involved development of the "barrier" concept; a formalism based on a set of sequential barriers to the release of radionuclides from the emplacement environment was adopted to compare rates of decay against rates of migration of radionuclides toward man. The current phase of the program is concerned with describing in detail the sediments, chemistry, and thermal and radiation environments that might affect emplacement and containment within the sea floor. Current studies are directed at problems of ion migration, mass transport, barrier breachment and short-circuit modes, and predictability of the environment, among others. Results to date indicate that in an undisturbed environment of pelagic red clays, ion migration will result from dispersion driven by concentration gradients, and that the clayey sediments have a high affinity for all the long-lived radionuclides of a waste mix. It was shown that a generic study site in the Central North Pacific has been stable for the past 70 million years, and is likely to remain so over the next one million years. To date, no technical reasons have been presented that precluded successful disposal of high-level wastes into submarine geologic media. (Auth) (LKH)

LABORATORY STUDIES; THEORETICAL STUDIES; SEA DISPOSAL; STORAGE, GEOLOGIC; RADIONUCLIDE MIGRATION; GEOCHEMISTRY; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; GEOLOGY; CLAYS; SEDIMENTS; CONTAINMENT; SITE SELECTION

<305>
Tamura, T., Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Plutonium Distribution in a Desert Pavement-Desert Mound Soil System in Area 11.

NVO-171; Environmental Plutonium on the Nevada Test Site and Environs, M.G. White, et al. (Eds.). U.S. Energy Research and Development Administration, Las Vegas, NV, (pp. 1-15).(1977, June)

Differences in the characteristics of the plutonium in the desert pavement and desert mound soil systems at the Nevada Test Site (NTS) are discussed. The mounds are formed by the filtering action of the vegetation as it intercepts saltating and creeping particles. Adjacent bare areas consisting of a gravelly surface typify desert pavement soils. For particle sizes less than 2 mm in desert pavement approximately 95% of the activity is found in the 0-2.5 cm surface layer. The remaining 5% is distributed in the lower horizons with indications of a higher concentration in the 7.5-10 cm layer. By comparison, the 0-2.5 cm surface layer of the desert mound contains about 60% of the activity with 30% and 7% in the 2.5-5.0 and 5.0-7.5 cm layers, respectively. The predominate grain size at the surface in a desert pavement area is gravel. However, in the desert mound the sand content decreases with depth from 93% at the surface to 77% in the 7.5-10 cm layer. In the same interval the silt content increases from 4 to 15% and the clay content increases from 0.5 to 3%. Density gradient segregation was used to further characterize plutonium distribution. In the desert pavement of Area 13 the plutonium was concentrated in the greater than 2.90 g/cm cm band, whereas in the desert mound it was concentrated in the 2.5-2.7 g/cm cm bands. In Area 11 the greatest plutonium distribution occurs in the 2.90 g/cm cm band. Evidence indicates that plutonium is moving downward into deeper horizons in the desert pavement soils. More samples will be taken and organic material will be included in the fractionation studies. (NDV)

Grain Size Distribution; Stratigraphic Unit Thickness

<305> CONT.
A DOSE; CLAYS; CONTAMINATION; DENSITY; PLUTONIUM; RADIONUCLIDE MIGRATION; SILTS; SOILS; GRAVELS; RADIOACTIVITY; DESERTS

<306>
Thornthwaite, C.R., J.R. Mather, and J.R. Bahamura, Laboratory of Climatology, Centerton, NJ.

Movement of Radiostrontium in Soils.

Science 131(3406):1675-1679. (1960, April 8)

Water plays a key role in moving Sr 90 through the environment by removing strontium from the atmosphere as precipitation and by redistributing strontium in the soil in gravitational water. The former is hard to estimate while the latter, through a climatic water-balance bookkeeping technique, is easier to quantify. According to this technique, when the amount of water exceeds the need of the vegetation and the soil is at its water-holding capacity a water surplus will exist. It is the water surplus which migrates downward through the soil as a leaching solution and is responsible for the redistribution of strontium in the soil. Experiments in a variety of soil types indicated a definite pattern of strontium movement in the soil responding to the various soil and water factors: a certain percentage of the original concentration moves downward to the next lower layer in the soil with every unit of water added. A downward movement depends on the cation-exchange capacity and pH of the soil and the leaching efficiency of the water solution. A mathematical model was set up to describe this pattern of strontium migration by using factorial and exponential functions. Tested against laboratory experiments, the model was successful. The model determined a pattern existed but the time necessary for establishment of such patterns depends on cation-exchange capacity of the soil, pH of the soil, leaching efficiency of the solution, and the volume of the leaching solution applied. A relationship defining leaching efficiency and its role is illustrated. The model was then compared to strontium movement in natural soils. Prediction of strontium migration was poor and this is in part due to lack of fallout values, errors in radiochemical analysis, and non-representative soil samples. (NPV)

ENVIRONMENTAL EXPOSURE PATHWAY; FALLOUT; FIELD STUDIES; GROUND WATER; LABORATORY STUDIES; LEACHING; MODELS, MATHEMATICAL; PRECIPITATION, METEOROLOGICAL; RADIONUCLIDE MIGRATION; SOILS; STRONTIUM

<307>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Environmental Effects from Current Operations of Transuranic Waste Management.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 5, (pp. 5-1 - 5-12), 506 pp. (1979, February)

The effects of current Radioactive Waste Management Complex (RWMC) operations on workers, the public, and the environment are minimal. There are no normal releases of TRU contamination. Four TRU waste handling accidents have been recorded at the RWMC since the present management procedures were established in 1970. Only one of these accidents resulted in a release of contamination. There were no significant radiation doses to personnel. Surface and subsurface water samples are routinely collected at the RWMC. The only significant TRU contamination detected in water samples was in 1962 and 1969, when local floodings occurred. The floodwaters came in contact with buried waste in partially filled pits and trenches. The highest alpha activity measured in the runoff water was 1 X 10(E-8) uCi/ml. TRU contamination has been detected up to 8200 ft from the RWMC. The maximum contamination levels found in surface soils (0-5 cm) were 2048 nCi Am 241/sq m, 1377 nCi Pu 239/sq m, and 32 nCi Pu 238/sq m. Subsurface migration of radionuclides from the buried TRU waste is being monitored. Samples indicate that migration to date is limited to a few inches at the soil-waste interface. The study of uptake of radionuclides by animals and plants was started recently. Low concentrations of Pu 238, Pu 239, and Am 241 were found in deer mice tissues. The RWMC is a controlled area; consequently the potential for direct radiation exposure to the public is minimized. (NAP)

WASTES, TRANSURANIC; ACCIDENTS; WASTE MANAGEMENT; WASTE DISPOSAL; CONTAMINATION; RADIATION DOSE; PERSONNEL; SURFACE WATERS; GROUND WATER; SAMPLES; MONITORING; FLOODS; BURIAL; PITS; TRENCHES; ALPHA PARTICLES; SOILS; RADIONUCLIDE MIGRATION; ANIMALS; UPTAKE; AMERICIUM 241; PLUTONIUM 239; PLUTONIUM 238; VEGETATION; DISPOSAL SITE

<308>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Environmental Effects of Alternatives for Managing Transuranic Waste.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 12, (pp. 12-1 - 12-81), 506 pp. (1979, February)

For each alternative and concept, the potential radiological and nonradiological effects on the environment at Idaho National Engineering Laboratory (INEL) were evaluated. These effects would derive from the following phases: site preparation and construction, plant operation, and decontamination and decommissioning after completion of the waste management project. Summary tables highlight some of the major nonradiological effects such as construction material usage, particulate emissions, labor, and diesel fuel usage. Minor and temporary impacts are expected in air quality and surface water quality during the construction phase. Increased noise levels from construction would be localized. Impacts on vegetation would be limited to removal of vegetation from the construction site. Habitat and cover for wildlife would be reduced (about 0.07% of the total acreage at INEL). Construction of waste management facilities would result in long-term or irreversible commitment of some natural

<308>

ENVIRONMENTAL TRANSPORT

<308> CONT.

resources. Added employment during construction or operation of any of the facilities would not have a major socioeconomic impact. For all alternatives and concepts, the radiological dose commitments to the general public from normal operational releases would be a very small fraction of the natural background dose. The whole-body dose from natural background radiation in southeastern Idaho is about 150 mrem/yr. For any of the alternatives, the maximum whole-body dose commitment to any individual, resulting from one year of normal operation, was calculated to be 1.9 X 10(E-7) mrem. The annual population dose commitments can be compared with the annual population doses from external and internal radiation from natural sources. In 1985, the population of about 136,000 within 50 miles would all receive approximately 1.5 X 10(E+2) mrem/yr from natural sources. The estimated population dose would be about 2 X 10(E+4) man-rem in that year. The largest whole-body population dose from pursuit of any waste management alternative would be about 2.8 X 10(E-6) man-rem, or 10 orders of magnitude lower than the population dose from natural sources. (RAF)

Numerous tables give data on radiological and nonradiological effects of alternatives and concepts studied.

WASTE MANAGEMENT; WASTES, TRANSURANIC; AIR QUALITY; DUSTS; ANIMALS; SOILS; EROSION; RUNOFF; WINDS; LAND USE; SURFACE WATERS; GROUND WATER; WASTES, NONRADIOACTIVE; ECONOMICS; DECONTAMINATION; DECOMMISSIONING; EXHAUST GASES; PARTICLES, AIRBORNE; ENVIRONMENTAL EXPOSURE PATHWAY; MAN; RETRIEVABILITY; PLUTONIUM 238; PLUTONIUM 239; PLUTONIUM 240; PLUTONIUM 241; PLUTONIUM 242; CURIUM 244; URANIUM 233; CONTAMINATION; SOILS; FALLOUT; RADIATION DOSE; DOSE COMMITMENTS; EXPOSURE, POPULATION; BACKGROUND RADIATION; TRANSPORTATION, RAIL; MONITORING; SITE PREPARATION; HAZARD ANALYSIS; VEGETATION

Projected effects from long-term release scenarios for each alternative were calculated based on lung doses from lava flow over the waste or a volcanic eruption up through the waste. The maximum estimated population dose is 4 X 10(E+5) man-rem. The uncertainties of the calculated doses and risks are estimated generally to be factors of one to two orders of magnitude, both upward and downward. The train crews in shipping the waste to the Federal Repository would comprise about 90 people. These people would receive a population dose of about 1 man-rem/yr. The general public along the route would receive a population dose of about 0.05 man-rem/yr. No short-term health effects to the public due to waste management operations and due to uncontrolled releases would be expected. Long-term effects (typically occurring several years after exposure) could occur. However, such effects, both somatic and genetic, would not be expected statistically. (RAF)

Summary tables list bone, lung and whole body doses due to accident effects of alternatives evaluated.

ACCIDENTS; HAZARD ANALYSIS; WASTES, TRANSURANIC; DOSE COMMITMENTS; MAN; EXPOSURE, POPULATION; DOSE RATE; RADIATION DOSE; EVALUATION; TIME FACTOR; RETRIEVABILITY; REPOSITORY; CRITICAL PATHWAY; BONES; LUNGS; WATER; ENVIRONMENTAL EXPOSURE PATHWAY; ALPHA PARTICLES; INHALATION; INGESTION; EXPOSURE, EXTERNAL; RESUSPENSION; FOOD CHAINS; DRINKING WATER; WASTE MANAGEMENT; WASTE DISPOSAL; PLUTONIUM 239; PLUTONIUM 238; PLUTONIUM 240; PLUTONIUM 241; PLUTONIUM 242; URANIUM 233; AMERICIUM 241; CURIUM 244; DAUGHTER PRODUCTS; WASTE STORAGE; EARTHQUAKES; VOLCANOES; EROSION; TORNADOES; LAVA; FIRES; BACKGROUND RADIATION; RADIATION EFFECTS; METEORITES; FLOODS; PRECIPITATION, METEOROLOGICAL; RADIONUCLIDE MIGRATION; AQUIFERS; WELLS; RIVERS; DAMS; LEACHING; GROUND WATER; EXCAVATION; SOILS; ICE; WASTE PROCESSING; TRANSPORTATION; CONTAINERS; INCINERATION; PACKAGING; EXPLOSIONS; COMPACTION; IMMOBILIZATION; SHAFTS; TUNNELS; DECONTAMINATION; DECOMMISSIONING; CARCINOGENS; GENETICS; MUTATIONS

<309>

U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Radiological Risks to the Public.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INWL Transuranic Waste, Section 13, (pp. 13-1 - 13-106), 586 pp. (1979, March)

Time periods considered in evaluating the radiological effects on the public from uncontrolled (accidental or nature-caused releases) of radionuclides were: 1) the period of up to 100 yr following implementation of an alternative, and 2) the post 100-yr period. The following potential causes of release were studied: process and handling accidents, natural disasters, and airplane crashes. The release consequences were calculated for all of the operational steps. The scenario of greatest importance in dose and risk from short-term releases is lava flow over the waste, leading to possible airborne release of a fraction of the radionuclides present. The operational scenarios of greatest importance are: 1) an explosion in the slagging pyrolysis facility, and 2) a fire in the retrieval facility. The population risk would be small compared with that from natural background radiation.

<310>

Wildung, R.E., and T.R. Garland, Battelle-Pacific Northwest Laboratories, Richland, WA.

The Relationship of Microbial Processes to the Fate and Behavior of Transuranic Elements in Soils, Plants and Animals.

PNL-2416; 85 pp. (1977, October)

The influence of soil physiochemical and microbial processes on the long-term solubility, form, and bio-availability of Pu and other transuranics important in the nuclear fuel cycle is reviewed. Soluble, diffusible Pu in soils (usually less than 0.1% of total) appears to be largely present as particulates of hydrated oxide, but evidence suggests that microorganisms may influence the solubility of Pu and that the non-particulate plant-available fraction is stabilized in solution by inorganic or organic ligands of limited concentration in soil. The role of soil microorganisms is discussed on the basis of the (1) known chemistry of organic ligands in soils, (2) effects on the soil microflora, and (3) principal microbial transformation mechanisms, including direct alteration

<310> CONT.

(valence state, alkylation), indirect alteration (metabolite interactions, influence on the physiochemical environment), and cycling processes (biological uptake and release on decomposition of tissues). The toxicity of Pu to microorganisms depends on Pu solubility in soil. However, soil microorganisms are generally resistant to Pu, with toxicity apparently due to radiation rather than chemical effects. High resistant bacteria, fungi, and actinomycetes have been isolated from soil, and these organisms have been shown to be capable of transporting Pu into the cell and altering its form in the cell and in solution. The resulting soluble Pu complexes tend to be of higher molecular weight than simple complexes (Pu-DTPA) and negatively charged. The form of Pu, although not well-defined, is dependent upon organism type, carbon source, and time of Pu exposure during growth. These factors, in turn, are a function of Pu source, soil properties, and soil environmental conditions. Preliminary studies indicate that Pu present in solution as organic complex is readily assimilated by the plant. Evidence to date indicates that organometal complexes serve mainly to deliver the metal to the root membrane, and the ligands are not taken up by the plant stoichiometrically with the metal. After passing the root membrane, Pu is transported to the shoots in the xylem through formation of a number of complexes with plant ligands. The form of Pu differs in leaves and stems, but greater than 90% was not present as the ionic Pu(IV) originally added to soil on which the plants were grown. A reevaluation of plant-to-animal transfer coefficients may be required as Pu incorporated in plant tissues is markedly more available to animals than Pu(IV) gavaged in the inorganic Pu solutions which have been used for previous measurements. Differences in gastrointestinal transfer of Pu in stems and leaves of alfalfa are related to differences in Pu solubility in these tissues. Thus, the form of Pu in soils and plants may be closely related to Pu availability to animals. (Auth) (PAF)

REVIEWS; MICROORGANISMS; TRANSURANICS; SOILS; ANIMALS; SOLUBILITY; PHYSICAL FORM; UPTAKE; CHEMICAL PROPERTIES; OXIDES; PARTICLES; TOXICITY; RADIATION EFFECTS; BACTERIA; FUNGI; ENVIRONMENTAL EXPOSURE PATHWAY; CELLS; COMPLEXES; ORGANIC COMPOUNDS; ROOTS; SHOOTS; LEAVES; STEMS; VALENCE; TRANSFER COEFFICIENTS; HUMIC ACIDS; PLUTONIUM 238; PLUTONIUM 239; AMERICIUM 241; AMERICIUM 243; CURIUM 242; CURIUM 243; CURIUM 244; NEPTUNIUM 237, VEGETATION

<311>

Yousef, Y.A., and E.F. Gloyna, Florida Technological University, College of Engineering, Orlando, FL; University of Texas, College of Engineering, Austin, TX.

A Transport Model for Long Term Release of Low-Level Radionuclide Solutions into a Stream Ecosystem.

Fate of Pollutants in the Air and Water Environments, I.H. Suffet (Ed.). Proceedings of the 165th National American Chemical Society Meeting, Philadelphia, PA, April 1975, John Wiley and Sons, New York, Part 2, (pp. 239-259).

Stream processes in flowing water bodies were simulated in small scale ecosystems. Problems encountered with the simulated system such as stratification, deposition, erosion, and relocation indicate the complexity of defining a real river system. Under continuous release of radionuclides, bottom sediments and plants concentrate radioactivity until such time as an equilibrium state is reached. Plants reach equilibrium much faster than sediments. The rate of uptake is directly proportional to the difference between the saturation concentration and the concentration in sediments or plants at a given point in time. Release of radioactivity back to solution was observed after cessation of the source. Under test conditions, cyclical diurnal uptake of Cs 134 by suspended phytoplankton in the flume was noted. The uptake followed a pattern similar to the phytosynthetic oxygen production and a linear correlation was found to exist. Also, suspended phytoplankton will tend to deposit on the plant leaves, stems, and bottom sediments. Therefore, the surface concentration by sorbing surfaces increases. (Auth)

MODELS; FRESHWATER SYSTEMS; STREAMS; RADIONUCLIDE MIGRATION; CESIUM 134; STRONTIUM 90; SEDIMENTATION; BIOTA; PLANKTON; DYES; DISPERSION; HYDRODYNAMICS; DIFFUSION; DILUTION; SORPTION; PRECIPITATION, CHEMICAL; UPTAKE; EQUATIONS; VEGETATION

<312>
Asher, J.H. (Ed.), Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

National Waste Terminal Storage Program Progress Report June 1977.

Y/OWI/TMP-3/9; 95 pp.(1977, November 1)

Geological studies are being made of the Appalachian and Michigan Basins, Gulf Coast salt domes, Permian Basin bedded salt, East Coast Triassic Shale Basin and Gulf Coast clays, volcanic rocks, and crystalline rocks. Research in argillaceous and crystalline materials and the cooperative field testing program in Sweden are carried out. Technical support projects include studies in the following areas: heat transfer/thermal analysis, waste/rock interactions, rock mechanics, borehole plugging, safety and reliability, shale and clay mineral studies, and data management. Ongoing work of facility projects is presented in five sections: engineering projects, repository projects, OWI facilities, generic preconceptual facility studies and spent fuel preconceptual facility design. Two sections of the program report are devoted to planning and analysis, and regulatory aspects. Regulatory aspects include environmental studies, hazard analyses, licensing requirements, and socio-economic impacts of repository construction and operation. Visitors, trips, reports generated, future activities, subcontracts negotiated, employment levels, and a OWI organization chart are listed in Appendix A through F. (NAP)

WASTES, RADIOACTIVE; WASTE STORAGE; REPOSITORY; DESIGN; GEOLOGY; HYDROLOGY; SALT DOMES; CORES; BOREHOLES; SHALES; SITE SELECTION; ROCKS; STRESS; DRILLING; FRACTURES; WATER; VENTILATION; MODELS, MATHEMATICAL; PERMEABILITY; THERMAL PROPERTIES; RADIONUCLIDES; SORPTION; RADIONUCLIDE MIGRATION; SPENT FUELS; ECONOMICS; SAFETY; ROCK MECHANICS; GEOCHEMISTRY; CREEP; HEAT TRANSFER; INSTRUMENTS; CEMENTS; CONCRETES; HAZARD ANALYSIS; CONTAINERS; CASKS; TRANSPORTATION; SOILS; LICENSING; LABORATORY STUDIES; FIELD STUDIES

<313>
Asher, J.H. (Ed.), Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

National Waste Terminal Storage Program Progress Report for Period October 1, 1976, to September 30, 1977.

Y/OWI-9; 403 pp.(1978, April)

Major task changes since the previous progress report include the following: 1) Studies of the geologic formations that underlie the Nevada Test Site (Eleana Shale and granite) and the Hanford Reservation (basalt). Also, preparations were made for these activities to be separated from the existing program and to be locally managed starting in FY 1978. The accelerated work at these two locations and the expected independent management reflect the ERDA perception of the prospective suitability of the underlying formations and the future acceptability of repositories at these locations. 2) At least two locations for a geologic repository in salt formations are to be selected and certified by the end of FY 1979. The specific states where work is now planned include Utah, Texas, Louisiana, Mississippi, Ohio, and New York. 3) One repository at one of the two salt locations is to be designed, constructed, and put into operation in 1985. 4) The site screening process and repository engineering design should be directed at the primary requirement of disposing of spent fuel elements with the option of incorporating long-term retrievability storage in an initial operational phase. 5) The site screening process and repository engineering design should also provide for the capability of disposing of solidified reprocessing waste in conjunction with, or as an alternative to, disposing of spent fuel elements as waste. 6) Investigation of the suitability of other formations for deep geologic disposal of radioactive waste will continue. These studies will not be correlated with any schedule for repository development. (NAP)

WASTES, RADIOACTIVE; WASTE MANAGEMENT; REPOSITORY; WASTE DISPOSAL; GEOLOGIC FORMATIONS; SALT DEPOSITS; SALT DOMES; SHALES; GRANITES; BASALTS; SITE SELECTION; DESIGN; SPENT FUELS; RETRIEVABILITY; REPROCESSING; WASTES, SOLID; HEAT TRANSFER; WASTE-ROCK INTERACTIONS; ROCK MECHANICS; SAFETY; HAZARD ANALYSIS; BOREHOLES; FIELD STUDIES; LABORATORY STUDIES; EQUIPMENT; MODELS; MONITORING; ENVIRONMENTAL IMPACT STATEMENTS

<314>
Bartlett, J.W., Battelle-Pacific Northwest Laboratories, Richland, WA.

Alternative Waste Management Systems.

BNWL-2126; Nuclear Waste Management and Transportation Quarterly Progress Report, April Through June 1976, R.H. Platt, Section 1.0, (pp. 1.1-1.7), 46 pp.(1976, October)

The objectives of this program are to identify and evaluate alternative systems for treatment, interim storage, and final disposition of commercial radioactive wastes. During this quarter the development of remote sensing techniques for regional reconnaissance continued. These techniques have possible applications to the selection of waste repository sites. A draft of an Energy Knowledge Test and a literature review of past attempts at communication of energy information were completed. A draft was completed of a topical report assessing public values as they pertain to complex technological decisions. Studies of the reconcentration phenomenon continued. This situation can occur following a leach incident at an underground nuclear waste disposal site for certain radionuclide decay products in particular disposal environments. As a result of the reconcentration effect, the predicted maximum decay product discharge rates to the biosphere can increase above their corresponding values for surface storage. Thus geologic isolation can increase the largest dose that any individual is projected to receive throughout geologic time above doses from surface storage even though it has significantly delayed the discharge of those nuclides to the biosphere. Conversely, doses averaged over all geologic time are decreased from surface storage values by geologic isolation. For example, as Th 230 moves through the geologic column, it continually decays to Ra 226, which moves

<314>

GENERAL STUDIES AND REVIEWS

<314> CONT.
increasingly ahead of its Th 230 parent because of its lower K value and hence faster migration. Thus Ra 226 created by decay of Th 230 released at a given time during the leaching period overtakes Th 230 which was leached from the disposal site earlier. Because this Th 230 continually creates additional Ra 226 from its own decay, the local Ra 226 concentration increases above the initial equilibrium value. If the path length is long enough for significant reconcentration to occur but short enough so that the reconcentrated Ra 226 does not decay before discharging to the biosphere, then the phenomenon becomes significant. Drafts of two papers describing these studies have been completed. (Auth) (LKN)

WASTE TREATMENT; WASTE STORAGE; STORAGE, INTERIM; WASTE DISPOSAL; WASTES, RADIOACTIVE; SITE SELECTION; METHODS; SOCIAL ASPECTS; RECONCENTRATION; RADIONUCLIDE MIGRATION; LEACHING; RADIOACTIVE DECAY; DAUGHTER PRODUCTS; STORAGE, GEOLOGIC; DOSE COMMITMENTS; THORIUM 230; RADIUM 226; THEORETICAL STUDIES

<315>
Bradley, D., Battelle-Pacific Northwest Laboratories, Richland, WA.

Overview of Waste Isolation Safety Assessment Program and Description of Source Term Characterization Task at Pacific Northwest Laboratories.

T/OWI/SUB-77/78260; W -te-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pg. 29-31), 92 pp. (1977, August 20)

The Waste Isolation Safety Assessment Program (WISAP) was initiated in FY 1977 at Battelle Pacific Northwest Laboratories to develop and illustrate the methods and obtain the data necessary to assess the safety of long-term disposal of high-level radioactive waste in geologic formations. Initially, the focus will be on generic isolation systems, but the methods and data will ultimately be applied to assessment of specific candidate sites. Activities of WISAP are divided into six tasks: (1) Safety Assessment Concepts and Methods, (2) Disruptive Event Analysis, (3) Source Characterization, (4) Transport Modeling, (5) Transport Data, and (6) Societal Acceptance. Task 3 of the program has the objective of determining the rate that radioactivity would be released from the waste form as a result of potential repository failure. Leach rate data will be obtained on all waste forms destined for geologic disposal such as high-level waste, spent fuel, cladding, low-level and TRU wastes. These waste forms will be extensively characterized prior to conducting leach experiments. Leach solutions representative of field conditions will be used and the effects of temperature, flow rate and surface alteration due to geologic storage will be investigated. The surface alteration effect will be studied by autoclave tests on mini-canisters in contact with repository rock and solution types. Currently high-level waste glasses are being studied in two series, one by using glasses doped to expected loadings of actinides, and the other using actual full-level waste glasses. High-level waste glass of Nuclear Fuels Services composition will be tested; upon final development of Allied General Nuclear or Exxon waste glasses, these may also be studied. The effect of devitrification on leach rate will be studied along with the effect of the absence of oxygen in the geologic storage media. (Auth) (LKN)

Only the abstract is given.

WASTE-ROCK INTERACTIONS; SAFETY; WASTES, HIGH-LEVEL; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; WASTE STORAGE; SITE SELECTION; HAZARD ANALYSIS; MODELS; RADIONUCLIDE MIGRATION; LEACHING; SPENT FUELS; CLADDING; WASTES, TRANSURANIC; WASTES, LOW-LEVEL; THERMAL PROPERTIES; HYDRAULIC VELOCITY; SURFACE PROPERTIES; GLASS; ACTINIDES; LABORATORY STUDIES

<316>
Bradshaw, R.L. (Ed.), and W.C. McClain (Ed.), Oak Ridge National Laboratory, Health Physics Division, Oak Ridge, TN.

Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines.

ORNL-4555 (Draft); ORNL-CF-70-7-42; 822 pp. (1971, April)

Project Salt Vault is the culmination of a research and development program aimed at establishing the feasibility, safety, and techniques for disposing of high-level power reactor wastes in natural salt formations. The background and early studies leading to the experiment are outlined in the first few chapters, followed by a large portion of the report detailing descriptions of the demonstration itself. Technical results obtained, including an exhaustive interpretation and analysis of their significance, are presented. It is concluded that most of the major technical problems pertinent to the disposal of highly radioactive wastes in salt have been resolved. Project Salt Vault successfully demonstrated the stability of the salt under the effects of heat and radiation as well as the capability of solvity mine structural problem by standard mining techniques. The data on the deformational characteristics of salt have made it possible to arrive at a suitable design for a mine disposal facility. The costs of this method of disposal in specially mined chambers in bedded salt deposits seems to be completely compatible with competitive nuclear power and only slightly more than perpetual tank storage. Each chapter has been input separately into the data base. (CAJ)

Numerous tables, graphs, illustrations, and references are given in each chapter.

WASTE MANAGEMENT; WASTE STORAGE; WASTES, SOLID; WASTES, RADIOACTIVE; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; GEOPHYSICAL SURVEYS; RECOMMENDATIONS; DISPOSAL SITE; DESIGN; MINES; RADIATION EFFECTS; REVIEWS; SITE EVALUATION; THERMAL PROPERTIES; COST BENEFIT ANALYSIS; MINING; EVALUATION; ROCK MECHANICS; LABORATORY STUDIES; FIELD STUDIES; INSTRUMENTATION; HAZARD ANALYSIS; TEMPERATURE

GENERAL STUDIES AND REVIEWS

<317>
Canadian Department of Energy, Mines and Resources, Canada.

The Management of Canada's Nuclear Wastes.

Atom 257:70-76. (1978, March)

A synopsis of a study of the Canadian nuclear program is presented. The study based its recommendations on the once-through fuel cycle concept typified by the CANDU reactor system. Wastes arising from this system are irradiated fuel elements and gaseous, liquid and solid wastes from reactor operation and maintenance. One of the recommendations the review group made was that no commercial fuel reprocessing plant should be approved in Canada until satisfactory methods of dealing with the wastes and waste products are demonstrated. The reprocessing would not eliminate the long-term hazard of plutonium and other actinides. The problems involved in stealing a significant quantity of irradiated fuel and using it to threaten a population were so immense that such an occurrence was unlikely. A storage bay being bombed would also not represent a significant hazard to the population. However, the security of separated plutonium is in doubt and quantities of such material should not be allowed in Canada. In regards to environmental and health impacts the group expected virtually no risks. However, a comprehensive environmental impact analysis is suggested. For disposal the group preferred crystalline rocks like those found in the Precambrian Canadian Shield. The possibilites of deep sea disposal require close watch. The group concluded that too little financial support and priority was given to the research and development program. More work in the areas of geology, geophysics, geochemistry, and engineering for repository sites is required. The final chapter of the report emphasized that the interim waste management techniques were being well developed in Canada and presented little hazard to the public. (NDV)

BEDROCK; CONTAINMENT; ENVIRONMENT; GEOLOGY; HAZARD ANALYSIS; HYDROLOGY; IGNEOUS ROCKS; GEOCHEMISTRY; POPULATIONS; REPOSITORY; RECOMMENDATIONS; SEA DISPOSAL; WASTE MANAGEMENT; WASTE DISPOSAL; WASTE STORAGE; FUEL REPROCESSING; WASTES, RADIOACTIVE; PLUTONIUM; ACTINIDES

<318>
Carruthers, J., PP Publications, Ottawa, Ontario, Canada.

Are Plutons the Answer to Nuclear Waste Disposal?.

Science Forum 8(6):15-16. (1975, December)

This article is a subjective discussion of Atomic Energy of Canada Limited's (AECL) recent change in policy with respect to storage of high-level radioactive waste, and its initiation of a joint investigation with the Geologic Survey of Canada to study salt deposits and granitic plutons as possible sites for long-term, permanent, underground storage. The plutons, which have been relatively stable for millions of years, have little mining interest, are plentiful throughout the Canadian shield, and need to be studied for faults or cracks that might allow escape of wastes and subsequent pollution. The need for storage facilities depends on Canada's decision whether or not to use reprocessing of spent fuel from CANDU reactors to extract plutonium for enrichment. The AECL estimates that large storage facilities will not be needed until the turn of the century. The AECL's policy of 1972 was a preference for carefully monitored above-ground storage that would be safe yet would allow retrievability. The change in policy to temporary above-ground holding with permanent underground storage comes from the assumption that the ultimate in retrievability, the above-ground storage facility, is good only as long as society remains responsible and responsive. When society becomes a problem, below-ground burial sites may be safer. The burial decision may be an indictment of a society that continues to ignore social and long-term energy costs. If alternative energy systems are used, we might not need to bury a problem that could be minimized in the first place. (Auth) (DCW)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; STORAGE, ABOVEGROUND; STORAGE, GEOLOGIC; STORAGE, INTERIM; WASTE STORAGE; WASTE DISPOSAL; SOCIAL ASPECTS; ECONOMICS; GEOLOGY; SALT DEPOSITS; IGNEOUS ROCKS; PLUTONS; FAULTS; SITE SELECTION; FUEL REPROCESSING; REACTORS, CANDU; PLUTONIUM; GRANITES

<319>
Claiborne, H.C., Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

The United States Program for the Safety Assessment of Geologic Disposal of Commercial Radioactive Wastes.

CONF-770565; Workshop on Risk Analysis and Geologic Modelling, Proceedings of a Workshop, Ispra, Italy, May 23-29, 1977, (17 pp.).(1977)

The safe disposal of commercial radioactive wastes in deep geologic formations is the goal of the National Waste Terminal Storage (NWTS) Program. This program is managed for the U.S. Energy Research and Development Administration (ERDA) by the Office of Waste Isolation (OWI), Union Carbide Corporation, Nuclear Division. A comprehensive safety assessment program has been established which will proceed on schedule consistent with the start-up of two waste repositories in late 1985. Safety assessment begins with selection of a disposal site - that is, all geologic and hydrologic factors must indicate long-term stability of the formation and prospective isolation of wastes from circulating ground waters for hundreds of thousands of years. The long-term stability of each site under thermal loading must then be demonstrated by sophisticated rock mechanics analyses. Therefore, it can be expected that the sites that are chosen will effectively isolate the waste for a very long period of time. However, to help provide answers on the mechanisms and consequences of an unlikely breach in the integrity of the repository, OWI is sponsoring a Waste Isolation Safety Assessment Program (WISAP) at the Battelle-Pacific Northwest Laboratories. The overall objective of this program is an assessment of the safety associated with the long-term disposal of high-level radioactive waste in a geologic formation. This objective will be achieved by developing methods and generating data

<319> CONT.
necessary to characterize the safety of generic geological waste disposal concepts, which are to be applied in the assessment of specific sites. It is expected that no one particular model will suffice. Both deterministic and probabilisic approaches will be used, and the entire spectrum of phenomena that could influence geologic isolation will be considered. Other long-term safety-related studies that complement WISAP are in progress, for example, borehole plugging, salt dissolution and salt transport in vertical boreholes. A repository will require licensing by the U.S. Regulatory Commission (NRC), and it is anticipated that the safety assessment program will provide the necessary safety analysis for the required licensing procedures. These requirements for licensing are in the process of being formulated by the NRC. (Auth)

SAFETY; HAZARD ANALYSIS; STORAGE, GEOLOGIC; WASTE DISPOSAL; WASTES, RADIOACTIVE; REPOSITORY; SITE SELECTION; GEOLOGY; HYDROLOGY; GEOLOGIC FORMATIONS; THERMAL PROPERTIES; WASTES, HIGH-LEVEL; BOREHOLES; SALT DEPOSITS; LICENSING; RADIONUCLIDE MIGRATION

<320>
Davidson, R.H., Tennessee Valley Authority, Knoxville, TN.

Tennessee Valley Authority Away-From-Reactor Spent Fuel Storage Studies.

CONF-790602; Proceedings of the 25th Annual Meeting of the American Nuclear Society, Atlanta, GA, June 3-8, 1979. (1979, June)

The Tennessee Valley Authority (TVA) has initiated a study of spent fuel storage alternatives to cope with the lack of reprocessing facilities for the forseeable future. Inputs from all levels of Government and the general public are being sought in the public participation program. TVA has engaged the Scientists' Institute for Public Information to assist in formulating and carrying out this new approach to public involvement. Results of the evaluation of alternatives available to TVA for management of spent fuel after existing on-site pools are filled should become clear as Phase I of the study nears completion at mid-year. It appears that permanent spent fuel disposition facilities cannot be constructed on a schedule which will eliminate the need for away-from-reactor engineered storage. It is therefore believed prudent to have firm plans ready to ensure that TVA reactors are not shut down for lack of storage space. (Auth) (LKM)

STORAGE, ABOVEGROUND; PONDS; HOLDING PONDS; RETENTION PONDS; SPENT FUELS; WASTE STORAGE; SOCIAL ASPECTS; WASTE MANAGEMENT; NUCLEAR FACILITIES; NUCLEAR POWER; STORAGE, INTERIM; THEORETICAL STUDIES

<321>
English, T.D. (Project Manager), Jet Propulsion Laboratory, California Institute of Technology, Pasadena, CA.

An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management. Volume 2.

JPL Publication 77-59, Volume 2; 235 pp. (1977, August 12)

This volume consists of seven appendices which summarize the study strategy used in the report; describe the EPA, ERDA, and NRC programs for high-level radioactive waste management; summarize major issues arising at six national meetings dealing with radioactive waste management; summarize the results of interviews with five noted authorities in the field; and compare the transuranics content of several alternative LWR fuel cycles. From the results of this research it is felt that the resources devoted to high-level waste management by NRC and EPA have been inadequate in view of the critical roles these agencies play in developing standards, criteria, and regulations. The budgets of these two agencies have recently been increased substantially, but it is questioned whether the resources available to these agencies are commensurate with their responsibilities. There appears to be a lack of an adequate platform for systematic discussion of the sufficiency of the scientific data base for geologic disposal. An important outcome of such a discussion would be the development of decisions as to what parameters should be monitored, and for how long, during the test phase of a specific repository site. In addition, there appears to be a need for improved coordination of the nation's high-level nuclear waste program. NRC is mandated to license both commercial and military high-level waste repositories; NRC has not decided whether to require either site selection review and/or construction licenses for these repositories. It is felt that assumption of these responsibilities by NRC could hasten the ultimate development of these repositories. In the area of pilot plant retrievable repositories for high-level waste, NRC plans to license commercial but not military pilot plants. Extending NRC's licensing authority to include military pilot plant repositories would eliminate this inconsistency. Also, it is felt that alternative fuel cycles should be systematically studied from the standpoint of reducing problems associated with waste disposal, proliferation, and safeguards. (Auth) (LKM)

WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; FUEL CYCLES; WASTES, TRANSURANIC; REACTORS, LIGHT WATER; STANDARDS, FEDERAL; ECONOMICS; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; LICENSING; DISPOSAL [illegible]; SITE SELECTION; REVIEWS; PLANTS, WASTE TREATMENT; WASTE TREATMENT; RECOMMENDATIONS

<322>
Hermansky, B.

Nuclear Power in Sweden.

Moderna Energie 22(9):383-389; OLS-78-180. (1976)

An overview of the Swedish nuclear program and its developments are presented. The Swedish government in 1975 approved a plan for 11 nuclear power plants to be functioning by 1980 and producing 8 GWe. Five of the plants were already in operation at the time of the decision. The decision to make such a commitment to nuclear energy was a result of

<322> CONT.

rising fossil fuel energy costs and the demand for electrical energy which was measured at 71 TWe in 1972. In the fifties and early sixties heavy water reactors were being considered. However, improvement cost of 40-50 million Swedish kronor (Skr) for the active core area caused abandonment of the plan and the consideration of the light-water reactor. Boiling-water and pressurized-water reactors are now in use. The government plan approved in 1975 calls for 350 million Skr to be spent on research and development of new reactors over the next three years. Safety research is also being done. A five year program for studying the disposal of radwastes and other problems associated with the final phase of the fuel cycle was instituted. The basic approach of this program is treatment and disposal on a domestic basis. Sweden has an adequate number of stable geologic formations for disposal of high-level radwastes and the Swedish Geological Survey is studying waste disposal in granite rock formations, including experimental tests at Studsvik. Sweden, in the years to come, expects to extract 300,000 tons of uranium from its rich deposits of low-grade uranium ores in shales. In recent years opposition by antinuclear groups has grown so that to better inform the people about nuclear energy an extensive campaign was begun by the government. (NDV)

COST BENEFIT ANALYSIS; ECONOMICS; EQUIPMENT; FUEL CYCLES; GEOLOGY; GRANITES; LEGISLATION; MINING; NUCLEAR POWER; REACTORS, LIGHT WATER; REACTORS, BOILING WATER; REACTORS, PRESSURIZED WATER; SAFETY; SHALES; URANIUM; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTE DISPOSAL; REVIEWS

<323>

Houser, B.L., C.F. Holoway, and D.G. Madewell, Oak Ridge National Laboratory, Oak Ridge, TN.

Guide to Radioactive Waste Management Literature.

ORNL-5226; 382 pp. (1977, October)

This guide has been compiled to serve scientists, engineers, administrators, legislators, and private citizens by directing them to sources of information on various aspects of radioactive waste management. References have been selected from about 6000 documents on waste management contained in the computerized information centers in Oak Ridge, TN. These documents were selected, examined, indexed, and abstracted between 1966-1976 by several knowledgeable indexers, principally at the Nuclear Safety Information Center. The selected references have been further indexed and classified into 12 categories. These categories and the number of documents included under each are as follows: Management - General, 419; Treatment, 245; Storage, 108; Disposal, 311; Transportation and Handling, 27; Alternative, 13; High-Level Waste, 125; Intermediate-Level Waste, 38; Low-Level Waste, 62; Fuel Reprocessing, 85; Tritium, 38; and Bibliographies, 26. The bibliographic part of this guide exists in computerized form in the Health Physics Information System and is available through the Oak Ridge Information Center Complex for searching from remote terminals. (Auth) (LKJ)

WASTE MANAGEMENT; WASTE DISPOSAL; WASTE STORAGE; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LOW-LEVEL; TRITIUM; BIBLIOGRAPHIES; REVIEWS; TRANSPORTATION; WASTES, RADIOACTIVE; FUEL REPROCESSING

<324>

Jet Propulsion Laboratory, California Institute of Technology, Pasadena, CA.

Description of the Environmental Protection Agency Program for Reprocessing and High Level Waste Management.

JPL Publication 77-59; An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management, Vol. 2, English, T.D. (Project Manager), Appendix B, (pp. B-1 - B-63), 235 pp. (1977, August 12)

This appendix describes the present status of programs and projects undertaken by the Environmental Protection Agency with respect to nuclear fuel reprocessing and HLW management. Background information on the Agency, including its overall mission and budget, (with a detailed budget analysis of radiation activities) is provided. Also included is a discussion of the tasks to be completed in FY 1977 and FY 1978. The authority of the EPA for radiation protection programs is implemented by the Office of Radiation Programs (ORP), whose goal is to minimize risk, in a cost-effective manner, from exposure to 1) nuclear energy applications, 2) naturally-occuring radioactive materials, 3) medical and occupational radiation, and 4) nonionizing radiation. The priority radioactive waste management tasks of ORP during FY 1977 are to 1) develop draft environmental criteria for radioactive waste, 2) prepare a technical support document for HLW disposal, and 3) complete a final fuel cycle standard. Follow-on activities tentatively planned for FY 1978 in the area of environmental standards include 1) preparation of an environmental standard for HLW disposal, 2) completion of a data base for Rn standards and Pu recycle standards, 3) development of a Technical Support Document for Pu and Th fuel cycle standards, and 4) publication of those standards. In the area of radiation protection, the only task planned for FY 1977 is the proposed publication of guidelines for Pu clean-up in soils. Activities tentatively scheduled for FY 1978 include: 1) publication of final guidance for Pu in soil, 2) publication of environmental criteria for radioactive waste classes, and 3) publication of protective radiation guides (PRG's) for food and water contaminated by particulate releases. In the area of environmental impact assessment, in addition to the on-going Environmental Radiation Ambient Monitoring System (ERAMS), other activities planned for FY 1977 include development and validation of a pathway model and a report on the air pathway exposure model validation. In FY 1978, ORP plans to publish an interim hydrogeological model. Environmental Impact Statements reviews are planned in FY 1977 for the Generic EIS on waste management, GEIS on mixed oxides, EIS on spent fuel storage, and EIS on the Exxon fuel reprocessing plants. In FY 1978, ORP expects to review federal activities in the areas of safeguards, reprocessing, and spent fuel storage. In the area of technology assessment, ORP plans for FY 1977 to publish an assessment of a radwaste disposal site in the New York, and to report on the transport of radwaste at

<324> CONT.
West Valley, New York. In FY 1978, ORP plans to look at 1) waste management and siting issues for decommissioning of radioactive facilities, 2) site selection and baseline monitoring for ocean dumping of radwastes, and 3) the technical analysis needed to analyze the Th fuel cycle. (Auth) (LKN)

RECYCLING; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; ECONOMICS; ENVIRONMENT; WASTE DISPOSAL; FUEL CYCLES; WASTE STORAGE; STANDARDS, FEDERAL; SOILS; ENVIRONMENTAL EXPOSURE PATHWAY; MODELS; ENVIRONMENTAL IMPACT STATEMENTS; HYDROLOGY; PLANTS, WASTE TREATMENT; SPENT FUELS; DISPOSAL SITE; SITE EVALUATION; RADIONUCLIDE MIGRATION; SOIL TRANSPORT; SITE SELECTION; MONITORING; SEA DISPOSAL; THORIUM; PLUTONIUM; RADON; WASTE TREATMENT

<325>
Jet Propulsion Laboratory, California Institute of Technology, Pasadena, CA.

Description of the Energy Research and Development Administration Technological Program for Radioactive Waste Management and Reprocessing.

JPL Publication 77-59; An analysis of the Back-End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management, Vol. 2, English, T.D. (Project Manager), Appendix C, (pp. C-1 - C-62), 235 pp. (1977, August 12)

The overall goal of ERDA's waste management and reprocessing operations is to provide for safe, efficient, and timely handling and processing of spent reactor fuel and the fuel cycle wastes. To this end ERDA has developed numerous programs and projects to examine those areas determined to have unanswered technical or regulatory questions. A gross indication of the interest in any subject can be obtained through the examination of budgetary trends. In the case of waste management and reprocessing programs, funding has increased approximately 100% in FY 1977 and 60% in FY 1978 (projected). This is in relation to the total ERDA budget increase of 35% and 30% respectively. Significant funding increases occurred in the areas of nuclear fuel cycle supportive operations (FY 76-77-78, over 100% per year), waste management commercial (FY 76-77, 600%; FY 77-78, 78%), long-term waste management ERDA (FY 76-77, 80%), and management of surplus contaminated facilities (FY 77-78, 97%). An analysis of the budgetary trends in combination with the program, project, and task descriptions provides a basis for evaluation of ERDA's program directions. There appear to be two significant areas of waste management that are not currently being addressed by ERDA. The first is the question of the possibility of not reprocessing the spent fuel rods and either storing them for an indefinite time, or disposing of them (the stowaway or throwaway alternatives). The specific aspects of the stowaway/throwaway option that are of growing concern are the long-term storage, packaging, and final disposition of the spent fuel rods. The second area ERDA does not appear to be investigating is management operations in the case of no plutonium recycling-specifically, what waste management operations would be affected by the inclusion of plutonium in the waste stream and how the potential hazard and the projected pathways to man would be affected. (Auth)

SPENT FUELS; WASTE TREATMENT; WASTE MANAGEMENT; FUEL CYCLES; WASTE STORAGE; WASTE DISPOSAL; PLUTONIUM; ENVIRONMENTAL EXPOSURE PATHWAY; RADIATION HAZARDS; RECOMMENDATIONS

<326>
Konzek, G.J., and C.E. Sample, Battelle-Pacific Northwest Laboratories, Richland, WA.

Decommissioning of Nuclear Facilities - An Annotated Bibliography.

NUREG/CR-0131; 525 pp. (1978, October)

The purpose of this bibliography is to compile pertinent, unclassified references relating to the decommissioning, including decontamination and disposal, of nuclear equipment and nuclear fuel cycle facilities. It contains the results of a literature review sponsored by the U.S. Nuclear Regulatory Commission (NRC) to establish an information source to aid researchers, scientists, and engineers currently engaged in decommissioning studies. This bibliography is intended for primary use by organizations in their planning and implementation of the eventual decommissioning of their own nuclear facilities. Abstracts are presented from references from U.S. literature or from world-wide literature, along with identification numbers (1. through 726.) that are cross-referenced to author, report number, and subject (keyword) indexes. Also, keywords follow each abstract in the text as a further aid to reader applicability to that specific reference. There is an alphabetical subject (keyword) index, an alphabetical list of authors, and a report number index (alphabetized), all cross-referenced to bibliography identification numbers. To further facilitate use of this bibliography, each document has been classified into one of 9 categories: Mining and Milling (29 references); Reactors (183 references); Nonreactor Fuel Cycle Facilities (94 references); Environment/Exposure Pathways/Health Physics/Radioactive Maintenance (62 references); Chemical, Electro-, Mechanical Decontamination Devices, Equipment, and Methods (192 references); Calculations/Decontamination Chemistry/Decontamination Studies/Radiation Damage Stedies (89 references); Packaging/Storage Sites/Transportation/Wastes (71 references); Design/Facilitation (32 references); and Other Pertinent Literature (24 references). Also included is a glossary of decommissioning terminology. (Auth) (NDV)

BIBLIOGRAPHIES; DECOMMISSIONING; DECONTAMINATION; EQUIPMENT; FUEL CYCLES; NUCLEAR FACILITIES; REACTORS; REVIEWS; MINING; MILLING; ENVIRONMENTAL EXPOSURE PATHWAY; METHODS; PACKAGING; TRANSPORTATION; DESIGN; DISPOSAL SITE; WASTE STORAGE; WASTE TREATMENT; WASTE MANAGEMENT; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, LIQUID; WASTES, COMMERCIAL; RADIATION HAZARDS

<327>
Lennemann, W.L., R.E. Parker, and P.J. West, International Atomic Energy Agency, Vienna, Austria.

<327> CONT.
Management of Radioactive Wastes.

Annals of Nuclear Energy 3:285-295. (1976)

There are several areas of current concern either bordering on or regarding radioactive waste management in the nuclear fuel cycle. These areas have local, regional and global aspects and require either immediate attention or preparation for future implications. Those areas requiring immediate attention are better management and control over mill tailings, a shortage of irradiated fuel storage space, and decommissioning criteria. Fuel storage should be recognized as a viable alternative to reprocessing for at least the next decade and suitable storage space incorporated in nuclear reactor facilities. Regulatory bodies should establish decommissioning criteria which them should be taken into account during the planning and design of a nuclear facility. Areas indicating future implications are removal and confinement of gaseous radionuclides, particularly tritium and krypton-85. There is now also some concern regarding carbon-14 which should be investigated. Resolving the high-level and alpha-bearing waste disposal question seems to be one of the most serious problems and highest priority objectives facing the nuclear industry today. Use of suitable geologic formations appears to be the only technology which will be available for the next several decades. The alternative is surface storage in engineered facilites which does not finally answer the question. Consequently, demonstrating the disposition of high-level and alpha-bearing wastes in suitable geologic formations should proceed without delay. International co-operation in the establishment and use of regional centers rather than having a proliferation of local facilities for fuel reprocessing should facilitate significantly the management of the resulting radioactive wastes. It also would reduce the hazards associated with having a large number of sources for radioactive releases. Nuclear power programs can come closer to maturity through international co-operation in resolving the waste disposal question and minimizing the number of facilities involved in fuel reprocessing and radioactive waste management. (Auth)

REVIEWS; WASTE MANAGEMENT; WASTE STORAGE; FUEL CYCLES; MILLS; TAILINGS; URANIUM; DECOMMISSIONING; NUCLEAR FACILITIES; GASES; RADIONUCLIDES; TRITIUM; KRYPTON 85; CARBON 14; WASTES, HIGH-LEVEL; WASTES, TRANSURANIC; ALPHA PARTICLES; WASTE DISPOSAL; GEOLOGIC FORMATIONS; STORAGE, ABOVEGROUND; STORAGE, GEOLOGIC; REPROCESSING; HAZARD ANALYSIS; COST BENEFIT ANALYSIS

<328>
Lomenick, T.F., and F.H. Empson, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN; Oak Ridge National Laboratory, Instrumentation and Controls Division, Oak Ridge, TN.

Geology and Description of the Lyons Mine.

ORNL-4555 (Draft); ORNL-CF-70-7-82; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch.5, (pp. 5.1-5.23), 822 pp.(1971, April)

This chapter introduces the Lyons Mine, the site selected for the demonstration experiment. A discussion of the geology, history, and general conditions at the mine is presented. It was concluded that a considerable amount of repair, replacement, renovation, and clean up were needed before installation of the experimental equipment could begin. (CAJ)

Photos, maps, schematic drawings and figures are included.

WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; GEOPHYSICAL SURVEYS; DISPOSAL SITE; DESIGN; MINES; FIELD STUDIES; WASTES, SOLID; GEOLOGY

<329>
Mayman, S.A., R.N. Barnes, J.E. Gale, and B.V. Sanford, Atomic Energy of Canada, Limited, Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, Canada; Ontario Hydro, Toronto, Ontario, Canada; Geological Survey of Canada, Ottawa, Ontario, Canada.

The Canadian Program for Storage and Disposal of Spent Fuel and High-Level Wastes.

CONF-760310; IAEA-SM-207/91; AECL-5597; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 1, (pp. 49-58). (1976)

Spent fuel from Canadian reactors is presently stored at the reactor sites, in water filled bays. However, Ontario Hydro is planning to have a central fuel storage facility in operation by 1985. One water-cooled and two air-cooled concepts are being considered for use at this facility. One of the air-cooled concepts--the concrete canister--is being tested by Atomic Energy of Canada Limited, and results to date are encouraging. Methods for disposal of high-level wastes from reprocessing plants are being developed as well. A mined cavity in plutonic crystalline rock is favored, but salt is a strong candidate and the Geological Survey of Canada is carefully evaluating all Canadian salt deposits. A small study in serpentinite is also in progress. A geologic disposal facility is expected to be available for commissioning and test emplacements by 1985. By the year 2000 it could be licensed to accept wastes on a routine production basis. This timing is compatible with the requirements of the Canadian nuclear industry. (Auth)

WASTE STORAGE; WASTE DISPOSAL; SPENT FUELS; WASTES, HIGH-LEVEL; COOLANTS; WATER; AIR; CONCRETES; CAVITIES; MINING; IGNEOUS ROCKS; GRANITES; STORAGE, GEOLOGIC; SALT DEPOSITS; REPOSITORY; STORAGE, ABOVEGROUND; STEELS; CONTAINERS; WASTE MANAGEMENT; TRANSPORTATION; METAMORPHIC ROCKS; MAPS; DRILLING; ROCKS; MINES; SITE SELECTION; GEOLOGY; GROUND WATER; FRACTURES; PERMEABILITY; EXPOSURE, OCCUPATIONAL; RETRIEVABILITY; DECOMMISSIONING; ECONOMICS

<330>
McCarthy, G.J. (Ed.), R.L. Schwoebel (Ed.), R.W. Potter, II, (Ed.), A.N. Friedman (Ed.), J.G.

<330>

GENERAL STUDIES AND REVIEWS

<330> CONT.
Moore (Ed.), H.C. Burkholder (Ed.), and W. Lutze (Ed.), Pennsylvania State University, Materials Research Laboratory, University Park, PA; Sandia Laboratories, Albuquerque, NM; U.S. Geological Survey, Menlo Park, CA; Argonne National Laboratory, Argonne, IL; Oak Ridge National Laboratory, Oak Ridge, TN; Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH; Hahn-Meitner-Institut, Berlin, German Federal Republic.

Scientific Basis for Nuclear Waste Management.

CONF-781121; Science Underlying Radioactive Waste Management, Vol. 1, Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY; 563 pp.(1979)

A symposium of waste management is presented. The major topics of discussion are solid waste forms, characterization of nuclear waste forms, existing management plans, radwastes in geologic settings, metallurgical considerations, radionuclide transport in geomedia, and modelling and safety assessment. Each abstract is reported separately. (NDV)

GLASS; GEOLOGY; RADIONUCLIDE MIGRATION; SAFETY; HAZARD ANALYSIS; WASTE MANAGEMENT; TRANSPORTATION; MODELS; SOLIDIFICATION; IMMOBILIZATION; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE

<331>
Morton, R.J., Oak Ridge National Laboratory, Health Physics Division, Radioactive Liquid Waste Research and Development Section, Oak Ridge, TN.

Waste Disposal Problems.

NASH-275; Sanitary Engineering, Proceedings of a Conference, Baltimore, MD, April 15-16, 1958, (pp. 320-329).(1958)

The nuclear energy industry is in a stage of expansion and technological development for peacetime uses. From the viewpoint of radioactive wastes this expansion and the accompanying developments may be expected to present new and perplexing problems of control and disposal; and potential hazards of much greater magnitude than have been considered in the past. This period of expansion presents the opportunity and the challenge for sanitary engineers to adapt their special knowledge and experience and to achieve leadership in developing the criteria and the methods that are essential for satisfactory radioactive waste disposal. (Auth)

BURIAL; CONTAINMENT; CONTAMINATION; DISPOSAL SITE; ENVIRONMENT; EVALUATION; FISSION PRODUCTS; HAZARDS; MONITORING; NUCLEAR POWER; WASTE DISPOSAL; WASTE MANAGEMENT; WASTE TREATMENT; WASTES, RADIOACTIVE

<332>
Nuclear Safety and the Environment.

Research and Development for Nuclear Power: 28-31.

Nuclear safety research is coordinated by the Safety and Reliability Directorate; projects being carried out include treatment studies of fuel element cladding, transuranic separations, nuclear transmutation, risk assessment for atmospheric dispersion of radionuclides, and techniques for compounding probabilities, improvement of pressure vessel inspection techniques, radiation physics and dosimetry studies, and development of a non-destructive waste drum Pu-monitoring system. Work is continuing on development of the HARVEST process for conversion of liquid radwaste to glass. Pilot plant trial runs have been made to identify and solve design requirements for feed and off gas treatment. Glass stability has been tested by dosing with 8000 year-equivalents of alpha radiation without visible effect. Leaching tests are in progress. A feasibility study is being carried out for disposal of vitrified radwaste into hard rock formations, and a tentative field study site in southwest Scotland has been selected. A new research program with Euratom support concerns the resuspension of heavy metal oxide particles as a function of windspeed, soil moisture, soil type, texture, and density of vegetation. A second project will determine the factors controlling tritiated water vapor interaction with natural surfaces. Soil cores and vegetation are being collected throughout the UK on a 100 X 100 km grid and analyzed for transuranics and Cs 137; measured Cs deposition over the last 17 years will be related to a reference population and used to assess future trends. (LKH)

SAFETY; REACTORS; COOLANTS; POLLUTION, AIR; GLASS; WASTES, LIQUID; LEACHING; STORAGE, GEOLOGIC; VOLUME REDUCTION; CLADDING; FUEL ELEMENTS; TRANSURANICS; SEPARATION PROCESSES; TRANSMUTATION; TRITIUM; SOILS; CORES; CESIUM; MONITORING; PLUTONIUM

<333>
Platt, A.M., Battelle-Pacific Northwest Laboratories, Richland, WA.

Nuclear Waste Management and Transportation Quarterly Progress Report, April Through June 1976.

BNWL-2126; 84 pp.(1976, October)

This document is one of a series of technical progress reports designed to report on commerical radioactive waste management programs at Battelle Pacific Northwest Laboratories. Programs covered under this report are Alternative Waste Management Systems; Decontamination and Densification of Chop-Leach Cladding Residues; Disposition (D and D) of Retired Contaminated Facilities at Hanford; Transportation Safety Studies; Transport Problems, 1976-2000; Safety and Economic Study of Special Trains; and Development of High-Level Waste Shipping Cask Models. The Alternative Waste Management Systems, Disposition of Retired Contaminated Facilities at Hanford, and Development of High-Level Waste Shipping Cask Models Programs have been abstracted separately. (LKH)

WASTE MANAGEMENT; DECONTAMINATION; DECOMMISSIONING; CLADDING; RESIDUES; TRANSPORTATION; SAFETY; ECONOMICS; TRANSPORTATION, RAIL; CASKS; DESIGN; MODELS; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, SOLID; RADIONUCLIDE MIGRATION; WASTE TREATMENT; SITE SELECTION; METHODS; SOCIAL ASPECTS;

<333> CONT.
RECONCENTRATION; THEORETICAL STUDIES; FIELD STUDIES; LABORATORY STUDIES

<334>
Powers, D.W., S.J. Lambert, S.E. Shaffer, L.R. Hill, and W.E. Weart, Sandia Laboratories, Albuquerque, NM.

Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico.

SAND-78-1596; 616 pp.(1978, December)

The Geological Characterization Report for the WIPP site presents, in one document, a compilation of geologic information available to August, 1978, which is judged to be relevant to studies for the WIPP. Chapters describe regional geology, site geology, seismology, hydrology, geochemistry, resources, special studies of WIPP repository rocks, and continuing studies. The chapters on regional and site geology, seismology, hydrology, geochemistry, and special studies have been abstracted separately. Chapter 8 discusses the amounts and types of resources likely to occur near the WIPP site rather than their present economic value. Potash and fluid hydrocarbons are the only two resources thought to be economically significant in the area. Caliche, salt, and gypsum are also present, but the abundance of these minerals throughout the region makes the effect of land withdrawal for the WIPP on their exploitation minimal. A number of geological characterization studies have not yet come to completion, or in some cases, have not begun. In regional geology, studies of paleoclimate, geomorphology, microfossils, and age-dating are planned to provide a history of past climate changes and to help assess the effects of future changes. Tectonic studies, including remote sensing and field studies, and seismic monitoring of structural features adjacent to the Delaware Basin will continue. Continuing studies of site geology include geologic mapping and high-resolution aeromagnetic surveys. Hydrologic studies will continue to expand the area around the site for which the groundwater behavior is characterized. Continuing studies in geochemistry are designed to support other investigations (such as the examination of dissolution products), provide detailed characterization of geologic materials to be used in in-situ and laboratory experimental programs (such as waste-rock interactions) and to contribute towards understanding of the nature of Ochoan (Permian) evaporites in general. No continuing studies of resources are planned. (LKH)

Extensive data are supplied in the form of maps, tables and graphs at the end of each chapter and in appendices at the end of the report.

GEOLOGY; HYDROLOGY; AGE DETERMINATION; TECTONICS; SEISMOLOGY; POTASH; GEOCHEMISTRY; WASTE-ROCK INTERACTIONS; EVAPORITES; WIPP; PETROLEUM; SITE EVALUATION; REVIEWS; STORAGE, GEOLOGIC; DEPOSITORY; SALT DEPOSITS; WASTE DISPOSAL; WASTE STORAGE; WASTE MANAGEMENT

<335>
Rockwell Hanford Operations, Richland, WA.

Rockwell Hanford Operations Quarterly Report, Process Technology and Process Development, July 1977 through September 1977.

RHO-LD-77-3 D; 149 pp.(1977, November)

This quarterly report is composed of summaries of technical progress on various investigations carried out at Hanford under Department of Energy sponsorship. These investigations cover such areas as waste management (commercial), separations, production, and scrap recovery, current waste management (defense), waste concentration, long-term management of defense waste, long-term management of contaminated soils and sediments, environmental impact technology, analytical methods technology development, and environmental technology. Under the National Waste Terminal Storage Program research is being conducted involving geologic mapping and geochemical analysis of basalts in the eastern Umtanum Ridge and Yakima Ridge-Rattlesnake Hills areas, core drilling and hydrologic studies, and waste-basalt compatibility studies. A report was completed discussing aspects to be considered in the design of a repository in the Pasco Basin. Waste fractionation studies to improve Sr and Cs recovery and purification process performance, waste concentration and encapsulation technology development, and waste processing, storage, and retrieval studies are among the areas reported on. (LKH)

WASTE MANAGEMENT; BASALTS; ENCAPSULATION; VOLUME REDUCTION; SALTS; SLUDGES; SITE SURVEILLANCE; WASTE STORAGE; TANKS; TRENCHES; RETRIEVABILITY; SAMPLING; WASTE DISPOSAL; ENVIRONMENTAL IMPACTS; SEDIMENTS; HYDROLOGY; RADIONUCLIDE MIGRATION; GROUND WATER; BIOSPHERE; GEOLOGY; SEISMOLOGY; HAZARD ANALYSIS; INSTRUMENTS; EQUIPMENT; FIELD STUDIES; LABORATORY STUDIES; RADIONUCLIDES

<336>
Steindler, M.J., H. Ader, R.E. Barletta, J.K. Bates, C.E. Bean, G.J. Bernstein, T.F. Cannon, R.A. Couture, K.F. Flynn, T.J. Gerding, L.J. Jardine, H. Krumpelt, B.J. Kullen, R.A. Leonard, W.J. Mecham, J.M. Moisenholder, K.M. Myles, R.H. Pelto, B.B. Saunders, W.D. Seefeldt, M.G. Seitz, A.A. Siczek, L.E. Trevorrow, S. Vogler, A.A. Ziegler, D.S. Webster, and L. Burris, Argonne National Laboratory, Chemical Engineering Division, Argonne, IL.

Chemical Engineering Division Fuel Cycle Programs Progress Report, January-March 1978.

ANL-78-68; 243 pp.(1979, April)

Fuel cycle studies reported for this period include studies of advanced solvent extraction techniques focussed on the development of centrifugal contactors for use in Purex processes. Miniature single-stage and eight-stage centrifugal contactors are being employed in studies of contactor performance and the kinetics of extraction. A 9-cm-ID centrifugal contactor has been completed, and fabrication drawings are being prepared for a plant-scale contractor. In other work, tricaprylmethyl-ammonium nitrate and di-n-amyl n-amylphosphonate are being evaluated as extractants in the Thorex process. In other investigation, literature on the dispersion of liquids by explosions is

<336> CONT.

being reviewed to develop a systematic and coherent body of knowledge as a basis for realistic source terms for accidents in fuel reprocessing plants. A process was developed for extracting TBP degradation products from TBP-Na2CO3 scrub solution while the actinides remain with the raffinate. In the program on pyrochemical and dry processing of nuclear fuel, the literature is being reviewed for acceptable materials for containment vessels; decladding methods are being evaluated; salt transport processes are being studied; a candidate flow sheet (based upon the Dow Aluminum Pyrometallurgical process) for reprocessing spent uranium metal fuel was prepared; work was begun on the use of molten salts for reprocessing actinide oxides; and the reprocessing of (Th,U)O2 solid solution in a KCl-LiCl salt containing ThCl4 and thorium chips was studied. Work on the encapsulation of solidified radioactive waste in a metal matrix includes study of (1) chemical interactions between simulated waste forms and matrix metals; (2) the leach rates of simulated encapsulated waste forms; and (3) the corrosion of candidate matrix metals and canister materials in brine solutions. Work to establish criteria for the handling of waste cladding hulls is continuing. The transport properties of nuclear waste in geologic media are being studied to estimate leaching of radionuclides from deep repositories by groundwater. The migration properties of water, iodine, and cesium in a column of kaolinite were also studied. The latter two studies are abstracted separately. (Auth) (LKH)

WASTE TREATMENT; WASTE PROCESSING; THOREX PROCESS; FUEL CYCLES; EXTRACTION; METHODS; PUREX PROCESS; ACCIDENTS; EXPLOSIONS; HAZARD ANALYSIS; PLANTS, WASTE TREATMENT; DECLADDING; CONTAINERS; CONTAINMENT; SALTS; ACTINIDES; FUEL REPROCESSING; ENCAPSULATION; WASTES, RADIOACTIVE; LEACHING; CORROSION; RADIONUCLIDE MIGRATION; STORAGE, GEOLOGIC; WASTE STORAGE; WASTE DISPOSAL; KAOLINITE; IODINE; CESIUM

<337>

Straub, C.P., University of Minnesota, Minneapolis, MN.

Radioactive Wastes.

Journal of Water Pollution Control Federation 47(6):1498-1513. (1975)

A review of the literature on radioactive wastes is presented. Data concerning sources and amounts of radioactive wastes are given along with predictions of future waste volumes. Waste treatment procedures discussed include adsorption or chemisorption of radioactive ions using a mixture of copper ferrocyanide and barium carbonate, and hydroxides of iron, titanium, and manganese; surfactants such as sodium dodecyl benzene and sodium dodecyl sulfate; evaporation procedures; and transmutation of long-lived radionuclides. General methods of waste storage are mentioned including storing high-level wastes as an aqueous solution, a gelatinous material, and a salt cake. Fixation methods are reviewed along with ground disposal in salt domes and disposal in the ocean. (JTE)

WASTES, RADIOACTIVE; WASTE DISPOSAL; WASTE TREATMENT; ADSORPTION; EVAPORATION; TRANSMUTATION; WASTE STORAGE; REVIEWS; RIVERS; PLUTONIUM; RADIOCHEMISTRY; RADIUM 226; REACTORS; SEAWATER; LAKES; ENVIRONMENT; PLANKTON; ION EXCHANGE; ECOSYSTEMS, AQUATIC; REPOSITORY; SALT DEPOSITS; SALT DOMES; STORAGE, GEOLOGIC; SEA DISPOSAL

<338>

Straub, C.P., University of Minnesota, Minneapolis, MN.

Radioactive Wastes.

Journal of Water Pollution Control Federation 44(6):1115-1123. (1972, June)

A book on all aspects of radioactive wastes was published by Rodier and Vernhes. A symposium on the developments in the management of low- and intermediate-level radwastes was held in France. Topics covered included standards for the control of effluents, nuclear power as an energy source, effluent control and monitoring, considerations affecting steam power station site selection, and benefit-risk assessments. The Transactions of the American Nuclear Society included summaries of 501 papers and about 250 references to published literature and reports produced by the Oak Ridge National Laboratory were listed. Improved processes on using activated carbon in a clarifier- contactor increased decontamination factors. Other processes employing precipitation, evaporation, and ion exchange were discussed. It was found that a two-step process of coagulation with the same consumption of iron sulfate increased decontamination of radwaste water by 12 to 20-fold and 2 to 5-fold with respect to alpha and beta activities, respectively. Consideration was given to a book on the theory of froth flotation of colloids as carriers of radioactive microcomponents and the processes and flotation methods. Solvent extraction was used to process irradiated nuclear fuels for Pm 147 as a mixture of lanthanides. Ion exchange by natural and synthetic materials was mentioned. The clay minerals were from White Oak Lake bed and the synthetic materials with some promise were phyllitic silicate minerals. Fixation by calcining and solidification in polyethylene or asphalt were discussed. Ground burial by hydraulic fracturing was found to have no significant present or future hazard to the public or the environment. Liquid wastes were placed in a deep underground chimney and allowed to self-boil. The rock refroze, trapping the radioactivity in an insoluble rock matrix. Burial in salt deposits is being further researched. No radwaste concentrations were reported in Milan, even though I 131, Hg 203, and Au 198 were disposed of in the sewers. Low-level Co 60 has been detected in localized areas of harbor bottom sediment around a few piers at operating bases and shipyards servicing vessels over a period of several years. The psychological impact on sea disposal was examined. Studies on the uptake by organisms of radwastes under current standards, guidelines, and operational experience indicated that and possible biological effects would be undetectable. (WDV)

ACTIVATED CARBON; ALPHA PARTICLES; ASPHALT; BETA PARTICLES; BURIAL; CALCINATION; CONTAMINATION; DECONTAMINATION; EVAPORATION; FLOCCULATION; HAZARD ANALYSIS; ION EXCHANGE; PRECIPITATION, CHEMICAL; STORAGE, GEOLOGIC; WASTE DISPOSAL; WASTE MANAGEMENT; WASTE STORAGE; WASTE

<338> CONT.
TREATMENT; WASTES, LIQUID; WASTES RADIOACTIVE; REVIEWS

<339>
Straub, C.P., University of Minnesota, Minneapolis, MN.

Public Health Implications of Radioactive Waste Releases.

World Health Organization Publication, Geneva, Switzerland; 61 pp. (1970)

The present volume is designed to provide assistance for health officers in the area of radiation contamination. Emphasis is placed on the health officer identifying the exposure pathways that are most important to the ecosystem of concern. To this end wastes are characterized, as well as sources, and collection, routes, and techniques of disposal are discussed. The wastes most likely encountered will be those released into the environment without further possibilities of control. Three techniques of management are available, they are: "concentrate and contain", "dilute and disperse", and "delay and decay". These methods are used depending upon the level of waste and to a small degree the environment. The sources of radwastes begin with ore mining and continue to the spent fuel and residual stage. Such sources may have pathways that are direct internal, indirect internal, and external. If an intelligent appraisal of possible hazards is to be made, sufficient samples in terms of numbers and types must be made. This will enable the identification of specific radionuclides, stable chemical substances, and other important constituents. Samples taken may be of liquid, gaseous, or solid forms, or a combination of the three. Examples are provided of specific routes and techniques of disposal of liquid and atmospheric discharge. Generally, a surveillance program of an establishment releasing radionuclides should include specific isotopic analysis for determining radionuclide levels. When transporting radioactive materials, consideration should be given to avoiding population centers. (NDV)

A general summary of waste production, disposal, and public health implications is presented.

ALPHA PARTICLES; BETA PARTICLES; HALF-LIFE, BIOLOGICAL; BODY BURDEN; BURIAL; CONTAINMENT; CONTAMINATION; CRITICAL PATHWAY; CROPS; DIFFUSION; DILUTION; DISPOSAL SITE; DOSE COMMITMENTS; DOSE RATE; ECOLOGY; ECOSYSTEMS, AQUATIC; ECOSYSTEMS, TERRESTRIAL; ENVIRONMENT; ENVIRONMENTAL EXPOSURE PATHWAY; ESTUARIES; RADIATION, GAMMA; GASES; GEOLOGY; GROUND WATER; HYDROLOGY; INCINERATION; INGESTION; INHALATION; ION EXCHANGE; LEACHING; LIVESTOCK; MAP; MAXIMUM PERMISSIBLE CONCENTRATION; MILLS; MINES; NUCLEAR FACILITIES; NUCLEAR MATERIALS; POLLUTION, AIR; POLLUTION, SOIL; POLLUTION, WATER; POWDS; RADIATION EFFECTS; RADIATION HAZARDS; HALF-LIFE, RADIOLOGICAL; RESINS; RIVERS; SLUDGES; TOXICITY; WASTE MANAGEMENT; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, SOLID; STANDARDS, INTERNATIONAL; STANDARDS, FEDERAL

<340>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored Idaho National Engineering Laboratory Transuranic Waste.

DOE/ET-0081; 596 pp. (1979, February)

The objective of the study is to provide preliminary evaluations of alternatives for long-term management of transuranic-contaminated (TRU) waste stored at the Radioactive Waste Management Complex (RWMC) at the Idaho National Engineering Laboratory (INEL). The study does not evaluate alternatives for buried TRU waste at INEL. A major portion of the national inventory of stored defense TRU waste is presently located at the RWMC. The TRU waste volume as of December 31, 1977, is approximately 1,200,000 cu ft is retrievable aboveground storage, and 2,000,000 cu ft is shallow-land burial. These volumes represent approximately 75% and 20%, respectively, of the total Department of Energy inventory. Additional TRU waste is continually placed into storage at the RWMC. A federal repository is required to alleviate the need for expanding the RWMC or establishing another waste location at INEL. Three possibilities are addressed with respect to the availability of a federal repository: 1) a federal repository will be available in 1985; 2) a federal repository will be available but will be delayed by 20 yr; and 3) a federal repository will not be available. The study uses existing data as well as data from ongoing projects. No experimental studies or other efforts were made to develop basic scientific data. The information given in the document includes: 1) the present environment of the INEL and the RWMC, 2) the RWMC and its operations, 3) the environmental effects from current operations, 4) the effects of current operations on workers and the public, and 5) the previous or current studies of TRU waste management. The long-term management alternatives are described in Sections 8, 9, and 10. Section 12 outlines the environmental effects while Section 13 evaluates the short- and long-term risks for the alternatives. Hazards to operating personnel are addressed in Section 14, and Section 15 examines the costs of alternatives. Implementation requirements, including land acquisition and use, necessary compliance documents, technical development studies and required environmental and safety studies are given in Section 16. Section 17 is a summary comparison of all alternatives studied. Fourteen sections were abstracted separately for the data base. (BAF)

WASTES, TRANSURANIC; WASTE STORAGE; REPOSITORY; EVALUATION; WASTE MANAGEMENT; RETRIEVABILITY; WASTE VOLUME; ENVIRONMENTAL EXPOSURE PATHWAY; MAP; HAZARD ANALYSIS; SAFETY; EXPOSURE, OCCUPATIONAL; COST BENEFIT ANALYSIS; LAND USE

<341>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Alternatives Involving Retrieval, Processing, and Shipment to the Federal Repository.

DOE/ET-0081; Environmental and Other Evaluations

<381>

GENERAL STUDIES AND REVIEWS

<381> CONT.
of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 9, (pp. 9-1 - 9-28). (1979, February)

This section discusses the methods studied for 1) retrieval of TRU waste stored at the Radioactive Waste Management Complex, 2) onsite processing, 3) packaging, and 4) shipment to the Federal Repository. Several options were considered for each of these operations, but only one option for each was evaluated further. The sequence of operations selected for study was retrieval with operator-controlled equipment inside a moveable, solid-frame structure at subatmospheric pressure; processing by slagging pyrolysis with the slag packaged in 55-gal drums; and rail shipment to the Federal Repository. Alternatives 3, 4, and 6 are discussed in this section. In Alternative 3, the TRU waste would be retrieved as scheduled (beginning in 1985 and requiring 10 yr), incinerated and immobilized by slagging pyrolysis, packaged, and shipped by rail to the Federal Repository. In Alternative 4, the TRU waste would be retrieved as scheduled (beginning in 1985 and requiring 10 yr), processed by slagging pyrolysis, packaged, stored in a precast concrete facility at the RWMC for 20 yr, and shipped to the (delayed) Federal Repository in 2005. In Alternative 6, retrieval would be delayed for 20 yr (to the year 2005). Processing would involve slagging pyrolysis and packaging. The TRU waste would then be shipped to the Federal Repository. There are significant uncertainties associated with planning for a 20 yr delay. If the ongoing monitoring and surveillance program for stored waste were to detect accelerated degradation of the waste containers, the delay of the retrieval for 20 yr would have to be reevaluated. (RAF)

Site plans for facilities for retrieval, onsite processing, and shipping to a Federal Repository are illustrated. Block flow diagrams show retrieval and processing steps. Plans are given of slagging pyrolysis facilities. A table shows flow rates of over-all system.

WASTES, TRANSURANIC; RETRIEVABILITY; WASTE PROCESSING; PACKAGING; TRANSPORTATION; REPOSITORY; WASTE STORAGE; WASTE MANAGEMENT; EQUIPMENT; COMPACTION; IMMOBILIZATION; INCINERATION; CONTAINERS; DRUMS; STEELS; WASTE VOLUME; GASES; DECONTAMINATION; DECOMMISSIONING; VENTILATION; THEORETICAL STUDIES; TRANSPORTATION, RAIL

<382>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Alternatives Involving Retrieval, Processing, and Disposal on the Idaho National Engineering Laboratory.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 10, (pp. 10-1 - 10-52), 586 pp. (1979, March)

In Alternative 5, the TRU waste would be retrieved from its present location, processed, and shipped to a disposal facility at the Idaho National Engineering Laboratory (INEL) site. Because there are three processing methods and four disposal methods, there are 12 possible combinations within this alternative. Retrieval would begin in 1985 or as soon thereafter as practicable. Three methods were analyzed for processing the stored waste: 1) incineration by slagging pyrolysis, followed by packaging; 2) compaction, immobilization, and packaging, 3) and packaging only. Onsite shipment of processed waste would be accomplished using semitrailers pulled by standard highway truck tractors. Four disposal methods were analyzed. In two of the methods, the processed waste would be placed in vaults or rooms mined out of the calcareous rocks in the Lemhi Mountain Range. These rocks are not part of the formation in which the Snake River Plain Aquifer is located. The two methods differ primarily in the access to the underground workings-shaft or tunnel. The third method involves engineered shallow-land disposal in lacustrine sediments at Site 14. Two design versions of this method were studied, affording different degrees of protection for the waste. The fourth method is disposal of the waste in an aboveground concrete structure near the Radioactive Waste Management Complex. For all methods and locations studied, the volumes, including containers, would be: 1) 1.3 million cu ft from the incineration process, 2) 1.9 million cu ft from the compaction and immobilization process, and 3) 3.4 million cu ft from the packaging-only process. (RAF)

Extensive illustrations are given of processing and disposal methods of Alternative 5.

WASTES, TRANSURANIC; RETRIEVABILITY; SITE SELECTION; WASTE PROCESSING; TRANSPORTATION, TRUCK; WASTE DISPOSAL; INCINERATION; PACKAGING; COMPACTION; IMMOBILIZATION; ROCKS; MINING; BASALTS; CLAYS; BENTONITES; SHAFTS; TUNNELS; SEDIMENTS; BURIAL; DESIGN; CONCRETES; WASTE VOLUME

<383>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Hazards to Workers.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 18, (pp. 18-1 - 18-18), 586 pp. (1979, February)

Both radiological and nonradiological hazards to the workers for each waste management alternative were evaluated on a preliminary basis. The nonradiological hazards evaluated were those leading to common industrial accidents such as falls, burns, and electric shock. The greatest number of injuries are associated with the alternatives that would involve retrieval and processing of the waste. Construction and operation of the slagging pyrolysis would contribute a major portion of such injuries. However, the number of injuries are no higher than those expected in similar industrial operations. Radiological hazards that were evaluated include the exposure of workers to radiation and possible contamination by radioactive materials. Hazards associated with normal operations and hazards associated with potential accidents were both considered. Radiological doses expected during normal operations would be far below the present occupational limit of 5 rem/yr. Largest exposures in man-rem would be expected in workers engaged in operations for waste retrieval and for slagging pyrolysis. Proper

<303> CONT.
facility design and operations, worker training, and health and safety monitoring would all contribute to performance of waste management functions with low level of occupational hazard. (RAF)

HAZARD ANALYSIS; SAFETY; EXPOSURE, OCCUPATIONAL; PERSONNEL; ACCIDENTS; ACCIDENTS, OCCUPATIONAL; WASTE MANAGEMENT; EXPLOSIONS, NON-NUCLEAR; EXPLOSIONS, NUCLEAR; FIRES; WASTE PROCESSING; TRANSPORTATION; INCINERATION; RADIATION HAZARDS; WASTES, TRANSURANIC

<304>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Requirements for Implementation.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 16, (pp. 16-1 - 16-6), 586 pp.(1979, March)

Implementation aspects discussed in this section are: land acquisition and use; regulatory requirements; technical development studies; and environmental and safety investigations. All of the Idaho National Engineering Laboratory disposal sites fall within the areas withdrawn from the public domain. Thus, there would be no requirement to purchase land, or to prepare or adopt special legislation. Several federal, state, and local agencies have cognizance over portions of the waste management activities. It is intended that the plans for siting, design, construction, operations and decommissioning facilities may be accomplished in accordance with all applicable regulations. Concepts developed during the present study were restricted to those based on state-of-the-art techniques and equipment. However, development work would be needed to adopt current technology to the envisioned application. Several environmental and safety investigations are underway or planned in support of the EIS for long-term TRU waste management. These investigations focus on waste characterization, container integrity, radionuclide migration and uptake, risk analysis studies, and ecological studies. (RAF)

LAND USE; LEGISLATION; STANDARDS, FEDERAL; STANDARDS, STATE; DISPOSAL SITE; WASTE MANAGEMENT; SITE SELECTION; DESIGN; DECOMMISSIONING; WASTES, TRANSURANIC; CONTAINERS; RADIONUCLIDE MIGRATION; UPTAKE; HAZARD ANALYSIS; SOILS; ANIMALS; ENVIRONMENTAL EXPOSURE PATHWAY; LABORATORY STUDIES; FIELD STUDIES; VEGETATION

<305>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Comparisons Among Alternatives, Concepts and Modules.

DOE/ET-0081; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 17, (pp. 17-1 - 17-18), 586 pp.(1979, March)

This section presents comparisons among the alternatives and concepts for long-term management of the stored TRU waste. Competing modules (e.g., slagging pyrolysis versus compaction and immobilization) are also compared. To quantify some of these comparisons, six general areas were selected: 1) radiological environmental effects, 2) short-term radiological risks, 3) long-term radiological consequences, 4) nonradiological hazards to workers, 5) radiological hazards to workers, and 6) cost. The evaluations performed in this document are preliminary, and not all the relevant factors for decision making are addressed. Comparisons of the alternatives on the bases of environmental effects and risk to the public are particularly difficult. The time periods for which these parameters were studied were 10 yr for Alternative 3, 30 yr for Alternatives 4 and 6, and many thousands of years for Alternatives 1, 2, and 5. To avoid misinterpretation of the data in an abstract, it is suggested that the reader refer to the document for a summary of conclusions reached. (RAF)

A table shows selected quantitative comparisons of alternatives and concepts.

WASTES, TRANSURANIC; EVALUATION; WASTE STORAGE; REPOSITORY; RADIATION HAZARDS; RADIATION EFFECTS; ENVIRONMENT; COST BENEFIT ANALYSIS; PERSONNEL; HAZARD ANALYSIS; RETRIEVABILITY; INCINERATION; WASTE DISPOSAL; WASTE PROCESSING; TRANSPORTATION

<306>
U.S. Energy Research and Development Administration.

Alternatives for Long-Term Management of Defense High-Level Radioactive Waste Idaho Chemical Processing Plant, Idaho Falls, Idaho.

ERDA-77-43; 170 pp.(1977, September)

Alternatives for long-term management of radioactive defense wastes located and generated at the Idaho Chemical Processing Plant (ICPP) are described and evaluated. The report describes the technology required for treating the waste, the risk to the public, and preliminary cost estimates. The present method used at the ICPP for handling high-level waste from fuel reprocessing operations converts the waste to a granular solid in a fluidized-bed calciner. This calcine has been retained in stainless steel bins within reinforced-concrete vaults since 1963. Compared to other waste forms, the calcine is a relatively leachable material; present development is directed towards converting the existing and future calcine to a less soluble and dispersible material. The alternatives considered for disposition of ICPP high-level waste consist of disposition locations and optional waste forms. The disposal locations alternatives are: 1) store solidified high-level waste at the ICPP in retrievable storage; 2) ship high-level waste to a federal repository; and 3) store nonactinide waste at the ICPP and ship the actinide waste to a federal repository. The calcined high-level waste can be converted to these optional forms: 1) pelletized calcine; 2) glass; 3) glass-ceramics; 4) sintered glass-ceramics; 5) metal matrix; and 6) concrete matrix. Any of these forms could be produced from actinide-free or actinide containing waste. Calculations show that

<306>

GENERAL STUDIES AND REVIEWS

<306> CONT.

risk to the public from implementation of any alternative is relatively small. The major radiation doses incurred are from routine releases, and the doses from these releases are well within established guidelines. The total population dose commitment from routine releases during processing would be 800 man-rem or less depending on the process. The maximum dose to an individual would be 1 mrem. By comparison, an individual receives about 165 mR/yr from background sources at this location in Idaho, and the total population background dose during the 10 yr that reprocessing is in progress is 200,000 man-rem. Preliminary costs for the alternatives were completed. The costs are based on preliminary designs or an actual costs for constructing similar systems at the Idaho National Engineering Laboratory (INEL). A cost-risk comparison shows that the existing method of treating the waste is the least expensive alternative based on both costs and risks. Because the risks of calcine disposal at INEL are negligible, all other alternatives result in higher risks and increased program costs. (HAP)

WASTES, HIGH-LEVEL; CALCINATION; WASTE STORAGE; REPROCESSING; CONTAINERS; STAINLESS STEELS; CONCRETES; WASTE MANAGEMENT; WASTE TREATMENT; EVALUATION; HAZARD ANALYSIS; COSTS; RETRIEVABILITY; TRANSPORTATION; REPOSITORY; ACTINIDES; PHYSICAL FORM; METALS; SINTERING; GLASS; CERAMICS; RADIATION DOSE; ACCIDENTS; CALCULATIONS; BACKGROUND RADIATION; DOSE COMMITMENTS; POPULATIONS; AIR

<307>

U.S. Energy Research and Development Administration.

Alternatives for Long-Term Management of Defense High-Level Radioactive Waste Hanford Reservations, Richland, Washington.

ERDA-77-44; 255 pp. (1977, September)

Alternatives that are being considered for the long-term management of defense high-level waste stored at Hanford in underground tanks and in stainless steel-lined concrete basins include placement of the waste in existing waste tanks, onsite engineered surface facilities, onsite geologic repository, and offsite geologic repository. The four major disposal alternatives are expanded into 27 alternative plans by considering: 1) variations in the final form of the high-level fraction (with radionuclide removal) to include glass, concrete and powder; 2) variations in the final form of the dehydrated waste product to include glass, calcined clay, and powder; and 3) variations in the treatment and handling of encapsulated waste to include packaging of capsules in canisters and conversion of the strontium fluoride and cesium chloride to glass. Canisters stored in sealed tanks on the surface are disposed of in a surface vault after the radionuclides have decayed sufficiently to avoid a heat transfer problem. A description of the technology, a preliminary risk assessment, and preliminary cost estimates for each of these 27 plans are presented. A 300-year period extending from 1990 to 2290 was used as the basis for the evaluation reported. General conclusions drawn as a result of the risk analysis are: Calculated radiological dose risks to individual members of the public [ranging from 2 X 10(E-12) rem/year to 3 X 10(E-3) rem/year] are extremely small compared to the dose due to natural background (100 millirem/year for the Hanford region). All of the alternatives are expensive to implement, but there are also large differences between them, particularly those involving disposal of in-tank waste. The cost ranges are greater for alternative plans requiring high-temperature processes and disposal in geologic repositories. The actual dollar amount that must be budgeted to implement any alternative plan is about two to five times the constant 1976 dollar amount. (HAP)

WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTE STORAGE; TANKS; CANISTERS; STAINLESS STEELS; CONCRETES; REPOSITORY; GLASS; CALCINE; CLAYS; ENCAPSULATION; STRONTIUM 90; CESIUM 137; FLUORIDES; CHLORIDES; CASKS; HAZARD ANALYSIS; COSTS; RADIATION DOSE; MAN; POPULATIONS; BACKGROUND RADIATION; AIR; ACCIDENTS; PARTICLES; GASES; RADIONUCLIDE MIGRATION; WASTES, LIQUID; SOILS; SALTS; SLUDGES; SOLIDIFICATION; WASTE VOLUME; RETRIEVABILITY; IMMOBILIZATION; WASTE STORAGE; WASTE DISPOSAL; FISSION PRODUCTS; ACTINIDES

<348>
A. Saberian and Associates.

National Waste Terminal Storage Review Meeting on Salt Dissolution.

CONF-770303; T/OUI/SWD-77/16537; Salt Dissolution, Proceedings of the National Waste Terminal Storage Program Review Meeting, Austin, TX, March 29-30, 1977, 38 pp.(1977, March)

The topics discussed are the salt dissolution in oil and gas test holes in Kansas, geohydrologic studies of Gulf Coast Intersalt domes, growth of a salt cavity by computer simulation, and salt transport in vertical boreholes and brine invasion of fresh water aquifers. Each one of these topics is abstracted separately. (NDV)

DISSOLUTION; AQUIFERS, CONFINED; SALT DOMES; SALT DEPOSITS; GEOLOGY; HYDROLOGY; CAVITIES; COMPUTER PROGRAMS; MODELS, MATHEMATICAL; NUMERICAL METHODS; BOREHOLES; CONVECTION; PREDICTIONS; LABORATORY STUDIES; FRESHWATER SYSTEMS; BRINES; REVIEWS

<349>
Anderson, E.L., F.G. Meyer, and P.K. Yeng, Rockwell International, Rocky Flats Plant, Golden, CO.

Fluidized Bed Incineration System for U.S. Department of Energy Defense Waste, July-December 1977.

RFP-2730; 6 pp. (1978, October 23)

A fluidized bed incineration is being evaluated at Rocky Flats Plant (RFP) as an alternative to convential incineration for processing transuranic combustible waste. The plant is designed for feed rate of 82 kg/hr which corresponds to a heat release rate of about 1.6 X 10(E+6) KJ/hr. This represents a scale factor of 9 based on solid waste feed rates. About 85% of the heat of combustion will be extracted through the walls of the afterburner by a water spray cooling system. A water cooled heat exchange in the flue gas stream will remove 55% of the heat from the combustion process. High speed blower will be used in the larger unit to provide active force for gas flow through the process. The four stage HEPA exhaust plane is undergoing final adjustments and is operational. As of November 1977 the equipment installation was 8% complete and completion is expected by February 1978. The process monitoring and control system diagnostic and operating software packages are 50% complete. (NDV)

ASHES; ACIDS; COMBUSTION; EXHAUST GASES; FLUIDIZATION; INCINERATION; GASES; GLOVE BOXES; PLANTS, WASTE TREATMENT; VOLUME REDUCTION; WASTE TREATMENT; WASTES, SOLID; WASTES, TRANSURANIC

<350>
Andersson, B., and P.A. Nalen, Stenlibergsgolagen, Ludvika, Sweden.

Mining Methods Used in the Underground Tunnels and Test Rooms at Stripa.

LBL-7041; SAC-08; 16 pp.(1978, August)

The area chosen for investigation was located away from the excavated part of the Stripa mine, towards the center of the massive granite body. The starting point and direction were based on the surface outcrops, geological maps, and cross sections of the mine. Two new methods, smooth blasting and slot drilling, were tested and used. The smooth blasting technique uses the symmetric pattern of the contour holes and a low charge density in each round, which results in a uniform and minimal damage to the roof and walls of the excavated cavern, and limits the number of newly opened fractures. The length of freshly opened fractures in meters is equal to the charge in kilograms per meter. The slot drilling technique uses an array of small-diameter peripheral percussion-drilled holes to drill ultra-large cores 1 meter in diameter and larger. Both techniques described have been successfully tested and their further use in future excavations of large storage caverns is recommended. (BAT)

MINING; METHODS; SITE SELECTION; DRILLING; EXCAVATION; FRACTURES; EXPLOSIONS; CORES; CAVITIES; STORAGE, GEOLOGIC; FIELD STUDIES; METHODS; FIELD STUDIES

<351>
Argino, E.L., University of Kansas, Lawrence, KS.

High-Level and Long-Lived Radioactive Waste Disposal.

Science 198(4320):885-890.(1977, December 2)

No uniform international policy of nuclear waste management exists. However, more and more countries are committing themselves to the use of nuclear power to generate electricity. Presently, all high-level wastes are being stored above ground in liquid form which is undesirable as a long-term solution. The other viable alternative to this storage is 100% recycling all wastes which is technologically feasible at present. Extraction processes remove 99.5% of Pu and U. If 99.9% removal for Pu, U, and Np and 99% for Sr and Cs is required, the long-term activity would be reduced by a factor of 100. With the actinides removed the waste problem time frame would be 700 to 1000 years. Three methods of burial are available: geologic, ice sheets, and seabed. Factors for selecting a rock formation are: wide distribution horizontally; high compressive strength, good thermal conductivity and high heat capacity; knowledge of groundwater hydrology; high tectonic stability; monomineralic or homogenous rock unit; and reasonable plasticity. The rock volumes needed to contain high-level radwastes are not large. By the year 2000 about 79 X 10(E+6) gal of waste will be produced in the US and if solidified would be reduced by factor of 8 and would occupy a football field to the height of 4 m. Regardless of the rock type detailed knowledge of heat capacity, thermal and radiation response must be had. Seabed disposal is a newer concept and several questions need to be answered. It is uncertain what the interaction of the sediments and heat will be and how stable the deep-sea environment is. Migration processes in the sediment and sea water-sediment boundary are not well enough understood to permit disposal. Not enough is known about the physics of an ice sheet or the ice free

<351> CONT.
areas in Antarctica to even begin an assessment on disposal in these areas. Another possible disposal area would be an arid zone. This concept has not been seriously considered and should be. (BCV)

A summary table of storage methods and technological feasibility for geologic, ice sheet, and seabed disposals are provided.

Waste Volume

ACTINIDES; ARGILLICEOUS; BEDROCK; BURIAL; CONTAINMENT; DISPOSAL SITE; ENVIRONMENT; GEOLOGIC STRATA; GEOLOGY; GROUND WATER; HYDROLOGY; RADIATION EFFECTS; ROCK MECHANICS; WASTE DISPOSAL; WASTE STORAGE; WASTES, SOLID; WASTES, LIQUID; WASTE VOLUME; SEA DISPOSAL; GLACIERS; THERMAL PROPERTIES; VOLUME REDUCTION; DECONTAMINATION FACTORS; REVIEWS

<352>
Angino, E.E., G. Dreschhoff, and E.J. Zeller, University of Kansas, Department of Geology, Lawrence, KS.

Antarctica - A Potential International Burial Area for High-Level Radioactive Wastes.

Bulletin of the International Association of Engineering Geology (14):173-178. (1976)

Disposal of solid high-level radioactive wastes in Antarctica can be considered through two separate approaches: burial in the massive ice sheet of East Antarctica; or, burial in bedrock in the dry valley or oasis areas of East Antarctica. Two techniques are possible in the disposal of high-level waste in ice sheets: the "melt-to-bottom" technique, and the "float-in-ice" or "tethered" technique. If thermally hot containers of solidified wastes are placed on the surface of the ice sheet, they will melt their own emplacement shafts, eventually coming to rest at the ice-rock interface. Snow, freezing of melt water, and subsequent plastic flow of the ice sheet would permanently seal the "shafts" behind the containers. A rough calculation of sinking rate would be about 1 m/day for a typical waste container. With an ice thickness of about 1.5 km, sinking time would be 3-5 years. Ice thickness in the proposed repository area is 1000 to over 3000 m; annual precipitation is less than 10 cm of water equivalent; average annual surface temperatures are -50 to -60 C; all of the area is more than 800 km from the continental edge. Preliminary considerations suggest that the best place to deposit the waste would be near the tops of ridges in the underlying bedrock. Paleoclimatological studies have shown that the continent has been glaciated at least 20 million years, and that average temperatures have not risen above 0 C for more than 1 million years. It is reasonable to conclude that the ice cap will not melt in the next 250,000-500,000 years. Although the likelihood of glacial surges as a result of thermal input from the waste is small, a thorough study is urged of the dynamics, structure, and heat budget of the East Antarctic ice cap and the sub-ice topography. An alternative which avoids this potential problem involves tethering waste canisters via a cable connected to a surface anchor. After 700-1000 years, heat generation would have dropped to a level where the canister would become frozen into the ice, without sinking to bedrock. The canisters would ride the ice sheet to the sea, arriving in an estimated 250,000-500,000 years. Advantages of this system are that wastes would be retrievable (with great difficulty), and a much smaller area would be required for the disposal system. A third approach would involve burial in the dry valleys or oasis areas; retrievability would be enhanced, the feasibility of dry storage is high, and problems of groundwater leaching are avoided (to a depth of at least 1 km groundwater is frozen). Problems with ice sheet disposal center on the lack of detailed information on ice sheet dynamics and heat transfer characteristics, and the difficulty of monitoring and evaluating disposal operations or of recovery from an unforeseen occurrence during transport to the disposal site. These objections are subject to engineering solution and are soluble with existing technology. (LKW)

WASTE STORAGE; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, SOLID; ICE; GLACIERS; STORAGE, GEOLOGIC; BURIAL; THEORETICAL STUDIES; RECOMMENDATIONS; SITE SELECTION; REPOSITORY; WASTES, RADIOACTIVE; GROUND WATER; THERMAL PROPERTIES

<353>
Bankhead, C.C., Wetherland, Sewell and Associates, Inc., Dallas, TX.

Geology of Certain Domes of the Northeast Texas Salt-Dome Basin.

Y/OWI/SUB-76/99939: Geologic Study of the Interior Salt Domes of Northeast Texas Salt-Dome Basin to Investigate Their Suitability for Possible Storage of Radioactive Waste Material, (pp. G-22 - G-57) 65 pp. (1976, May)

Six domes of the Northeast Texas salt-dome basin have been studied for the purpose of evaluating their tectonic and hydrologic stabilities. These domes are the Bullard, Keechi, Mount Sylvan, Palestine, Steen, and Whitehouse, located in Smith and Anderson Counties, Texas. The studies included an investigation of (1) the depth, size, and configuration of the salt mass, (2) the uplifting and warping of the sedimentary beds as the salt mass moved upward, (3) the position and thickness of the caprock associated with most of these domes, (4) the freshwater zones in the area and the hydrologic stability of the domes, and (5) the potential of the area for oil and gas development. A field trip was made to each of these domes to observe the surface geology, terrain, drainage patterns, and land use. Detailed descriptions of findings and maps and cross-sections are presented for each of the domes. A brief review of other domes in the basin is also included. Based on these studies, the Bullard, Whitehouse, and Keechi domes appear to have the highest degree of hydrologic stability, and therefore are the most suitable for further examination. However, additional data are needed to delineate more accurately the configuration of the salt mass, to examine the caprock, and to evaluate more completely the hydrologic stability of the domes. (Auth) (LKW)

SALT DOMES; HYDROLOGY; TECTONICS; GEOLOGIC STRUCTURES; GROUND WATER; AQUIFERS; GEOLOGY; LAND USE; STRATIGRAPHY; DISSOLUTION; CAP ROCK; STRUCTURAL STABILITY; FIELD STUDIES; REVIEWS;

<353> CONT.
SITE SELECTION; REPOSITORY; SEDIMENTARY ROCKS

<354>
Barney, G.S., Rockwell Hanford Operations, Richland, WA.

Variables Affecting Sorption and Transport of Radionuclides in Hanford Subsoils.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 835-838), 536 pp. (1979)

The identification and the derivation of a quantitative relationship between the variables and sorption parameters most important in the transport of radionuclides on sediments underlying the Hanford Reservation was undertaken. Statistically designed screening experiments were used to determine the most important factors. These factors were then studied in more detail using a response surface experimental design. This yielded regression equations relating equilibrium distribution coefficients for several significantly hazardous isotopes to the factors affecting sorption. The radionuclides considered are Cs 137, Sr 85, Tc 99, Ru 106, Co 60. For Cs 137 the significant variables affecting sorption are K^+, Na^+, and sediment source; for Sr 85 the variables are Ca^{2+}, Na^+, Mg^{2+}; for Ru 106 the variables are NO_2^-, and Ca^{2+}; and for Co 60 the variables are NO_2^-, and K^+. None of the variables had a detectable effect on Tc 99 sorption. There appears to be a correlation between surface areas of the sediments and Kd. The sediments with the largest surface areas also had the highest Kd values. (NDV)

DISTRIBUTION COEFFICIENT; SORPTION; STATISTICS; EQUATIONS; SOILS; SEDIMENTS; LABORATORY STUDIES

<355>
Helter, H.G., U.S. Atomic Energy Commission, Office of Environmental Affairs, Washington, D.C..

Deep Disposal Systems for Radioactive Wastes.

Memoir 18; Underground Waste Management and Environmental Implications, T.D. Cook (Ed.), Proceedings of a Symposium, Houston, TX, December 6-9, 1971. American Association of Petroleum Geologists, Tulsa, OK, (pp. 381-358), 612 pp. (1972)

As of July 1, 1971, approximately 100 million kw of nuclear-power-plant capacity was in operation or under construction and/or contract. During the 1960s, an extensive research, development, and demonstration program was carried out on the treatment and disposal of all types of gaseous, liquid, and solid radioactive wastes. Geochemical research and extensive field exploration and demonstration studies have been carried out on several deep disposal systems for radioactive wastes, including the application of hydrofracturing techniques in bedded shale for low-heat-producing wastes and the use of bedded salt and crystalline bedrock for highly radioactive wastes. The Atomic Energy Commission has adopted a regulatory policy which requires that all high-level liquid wastes from licensed irradiated-fuel-reprocessing plants must be solidified and shipped to a national repository on land owned and controlled by the federal government. A tentative selection of a site near Lyons, Kansas, has been made for an initial salt-mine repository for the demonstration of long-term storage for both solid high-level and long-lived alpha-contaminated wastes. Because of a general requirement for adequate monitoring to assure the safe and effective operation of a deep-well injection system, this method has not been used generally for disposal of radioactive wastes. It appears that injection into deep permeable formations may be a practical solution for the disposal of large quantities of tritium-bearing wastes from water reactors and nuclear-fuel-reprocessing plants in the future. Additional research is also required on the potential deep disposal of noble gases such as krypton 85 from reactor and reprocessing-plant off-gas streams. (Auth)

REVIEWS; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; SHALES; ROCKS; SALT DEPOSITS; BASALTS; BEDROCK; WASTES, SOLID; WASTES, HIGH-LEVEL; SOLIDIFICATION; TRANSPORTATION; LICENSING; SITE SELECTION; ALPHA PARTICLES; MONITORING; WELLS, INJECTION; WASTE DISPOSAL; TRITIUM; KRYPTON 85; GASES; XENON; HYDRAULIC FRACTURING

<356>
Bishop, W.P., D.R. Anderson, and C.D. Hollister, Sandia Laboratories, Albuquerque, NM; Woods Hole Oceanographic Institution, Woods Hole, MA.

Nuclear Wastes in the Deep Seabed.

SAND-76-7017; 8 pp.; CONF-750522; Second Biennial Topical Meeting on the Ocean, Nuclear Energy and Man, Tarpon Springs, FL, May 7-9, 1975. (1976)

The great mid-plate/gyre regions of the sea floor may provide a suitable resting place for transuranic high-level wastes presently predicted from the nuclear power industry. These regions, comprising about a quarter of the world's surface, are characterized by a lack of major currents, low biological productivity, thin sediments, low sedimentation rates, extremely low regional topographic gradients, lack of observed seismic events, great depth, and only one identifyable benthic resource. Because containment is the ultimate goal of disposal, a set of conceptual barriers against release from the seabed has been defined with some "best-guess" estimates of the number of years over which they may provide containment of the radionuclides. Conceptual barriers briefly discussed are: container type, waste form, rock type, sediment column, sediment surface, water column, plugging, accretion, and thermal effects. (SAP)

REVIEWS; WASTES, HIGH-LEVEL; WASTES, TRANSURANIC; SEA DISPOSAL; SITE SELECTION; SEDIMENTS; CONTAINMENT; RADIONUCLIDES; CONTAINERS; ROCKS; PLUGGING; ACCRETION; THERMAL EFFECTS; THEORETICAL STUDIES; LEACHING; BASALTS; SORPTION; ION EXCHANGE; PLUTONIUM; RADIONUCLIDE MIGRATION

<357>
Bogorov, V.G., and E.M. Krepa.

Concerning the Possibility of Disposing of Radioactive Waste in Ocean Trenches.

Peaceful Uses of Atomic Energy, Proceedings of the 2nd United Nations International Conference, Geneva, Switzerland, September 1-13, 1958. United Nations, Geneva, Switzerland, Vol. 18, (pp. 371-375). (1958)

Proposals that packaged high-level radioactive waste be disposed of in oceanic trenches are rejected on the basis of evidence that waters in trenches mix with overlying waters, creating the potential for radionuclide transport. In the case of the Tonga trench, maximum water density is 27.86 g/cc at -8000 m. below this depth, density decreases, indicating that mixing may be taking place. Measurements of temperature and salinity taken by the "Galatea" expedition in 1952 and the "Vityaz" expedition in 1957 show that relatively short-term changes take place. Temperature at 7 km changed 0.16 degrees C in 5 years, 0.11 degrees C at 8 km, and 0.16 degrees C at 9 km. Abundant oxygen below 3000 m (49-58% saturation) is the result of influx of Antarctic waters; phosphate levels are essentially constant below 1000 m (70 mg P/cum). Bottom sediments from the Tonga trench have a positive redox potential of 181 mV, oxidized iron compounds, and sulfates; the oxidizing layer is 15 cm thick, and at least 50 cm thick in the Kermadec trench. The data indicate significant mixing of oxygenated, relatively low-density, organically active water at depth in the Tonga trench and other trenches. (LKH)

Density

SEA DISPOSAL; TRENCHES; TEMPERATURE; SALINITY; OXIDATION; OCEANOGRAPHY; SEAWATER; SEDIMENTS; WASTE DISPOSAL; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; PHOSPHATES

<358>
Bradley, D.J., Battelle-Pacific Northwest Laboratories, Richland, WA.

Leaching of Fully Radioactive High-Level Waste Glass.

PNL-2658; 86 pp. (1978, September)

This report presents the results of leach testing of fully-radioactive high-level waste glass developed by PNL. Waste was generated by separating out the fission product and actinide fractions from power reactor fuel irradiated to 55,000 MWd/MTU; the liquid waste was batch calcined, combined with glass frit, melted, and formed into 100 g batches. Leach rates for Cs, Sr, Co, Sb, Mn, Pu, Ru, Rh, Cs, and Ce in deionized water, measured for 639 days, ranged from 4 X 10(E-5) g of glass/sq cm-day for Cs to 9 X 10(E-9) g of glass/sq cm-day for Ce. Average fractions released after 639 days were 2.8 X 10(E-2) for Cs, 2.0 X 10(E-2) for Sr, 1.6 X 10(E-2) for Co, 1.1 X 10(E-2) for Sb, 7.7 X 10(E-3) for Mn, 2.3 X 10(E-3) for Pu, 2.6 X 10(E-4) for Ru, 1.9 X 10(E-4) for Cs, and 8.8 X 10(E-5) for Ce. Another leach test, using the same glass composition, but with nonradioactive elements, yielded similar results for Cs and Sr. The data indicate that radionuclide release is initially by a diffusion-type mechanism, changing gradually to a silicate lattice alteration mechanism. Leach rates measured for samples from various portions of the glass melt indicated a homogeneous product. Further tests, involving geologic storage conditions and using fully radioactive wastes, are planned. (LKH)

Leaching data and isotopic composition of spent reactor fuel as calculated by an ORIGEN code computer run are included as appendices.

Leaching Rate

Mn 54; Co 60; Sr 90; Rh 106; Sb 125; Cs 134; Cs 137; Ce 144; Eu 154; Eu 155; Pu 239; Pu 240; Cm 242; Cm 244

VITRIFICATION; GLASS; WASTES, HIGH-LEVEL; LEACHING; RADIONUCLIDES; LABORATORY STUDIES; FISSION PRODUCTS; ACTINIDES; WASTES, LIQUID; CALCINATION

<359>
Bradshaw, R.L., W.J. Boegly, Jr., F.H. Empson, H. Kubota, F.L. Parker, J.J. Perona, and R.C. Struxness, Oak Ridge National Laboratory, Analytical Chemistry Division, Oak Ridge, TN; Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Ultimate Storage of High-Level Waste Solids and Liquids in Salt Formations.

Treatment and Storage of High-Level Radioactive Wastes, Proceedings of a Symposium, Vienna, Austria, October 8-12, 1962. International Atomic Energy Agency, Vienna, (pp. 153-175). (1963)

Salt formations offer an attractive site for the ultimate storage of high-level liquid and solid wastes. Salt formations are impermeable, have good structural strength and good thermal conductivity. Laboratory experiments and field tests in a Kansas mine have shown that: (1) the thermal conductivity and diffusivity of impure rock salt in situ (0.011 cal/cm/sec/ C and 0.026 cm(E+2)/sec at a peak salt temperature of 60 C) are within 10% to 20% of the laboratory values for single crystals; (2) elevating the temperature of 3000 gal (US) of waste in a 7 1/2-ft square cavity of 65 C did not upset the structural stability of the mine room, even though the floor-to-ceiling dimensions decreased by 1/2 inch at a distance of 5 ft from the cavity; (3) evaporation of solution with subsequent condensation on the walls produces some alteration of the cavity; (4) adsorption of the waste solution on the crushed salt appears to retard both cavity alteration and the production of radiolytic decomposition gases; (5) radiation does to be expected in an actual disposal operation will not be great enough to significantly affect the structural stability of the rooms. The major remaining uncertainty is the effect on structural stability of elevating the temperatures of large areas of a mine. A theoretical and experimental investigation of this problem is currently under way. Temperature-rise calculations have been performed for the storage of wastes from a 6-t/d fuel-processing plant. For liquids the waste was assumed to be absorbed on the granular salt in rooms with recessed floors. The net mine area required to limit the waste-temperature rise to 70 C varied from 34 acres per year for 120-day-old wastes to 5

<359> CONT.
acres per year for 30 year old wastes. If the wastes are stored in solid form above the mine floor, the average requirements are reduced by a factor 2.3. For cylindrical vessels stored in holes in the mine floor, calculations, based on an infinite array of the line source, showed that, in general, for wastes cooled longer than three years the space requirements are the same as for solids above the floor. Current studies are primarily concerned with the disposal of packaged solids and will culminate with a demonstration in a salt mine, probably during 1965. (Auth)

Thermal Conductivity; Plasticity

STORAGE, GEOLOGIC; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, SOLID; REPOSITORY, SALT DOMES; MINES; ABSORPTION; LABORATORY STUDIES; FIELD STUDIES; THERMAL PROPERTIES; EVAPORATION; GASES; RADIATION DOSE; CAVITIES; STRUCTURAL STABILITY; CONTAINERS; PLASTICITY; ECONOMICS; SITE SELECTION

<360>
Bradshaw, R.L., W.J. Boegly, Jr., F.H. Empson, N. Kubota, P.L. Parker, J.J. Perona, and E.G. Struxness, Oak Ridge National Laboratory, Oak Ridge, TN.

Ultimate Storage of High-Level Waste Solids and Liquids in Salt Formations.

Retention and Migration of Radioactive Ions in Soils, Proceedings of a Conference, Saclay, France, October 16-18, 1962 (pp. 257-268).(1962, October)

Laboratory and field studies in a Kansas salt mine have shown that: 1) the thermal conductivity and diffusivity of impure rock salt in situ (0.011 cal/cm/sec/deg C and 0.026 sq cm/sec at peak salt temperature of 60 C) are within 10% to 20% of the laboratory values for single crystals; 2) elevating the temperature of 3000 gal of waste in a 7 1/2 ft square cavity to 65 C did not upset the structural stability of the mine room, even though the floor to ceiling dimensions decreased by 1/2 inch at a distance of 5 ft from the cavity; 3) evaporation of solution with subsequent condensation on the walls produces some alteration of the cavity; 4) absorption of the waste solution onto crushed salt appears to retard both cavity alteration and the production of radiolytic decomposition gases; 5) radiation doses to be expected in an actual disposal operation will not be great enough to significantly affect the structural stability of the rooms. Temperature rise calculations have been performed for the storage of wastes from a 6 ton/day fuel reprocessing plant. Also calculated were approximate cost figures for a mine operation handling the entire United States' high-level waste production in the 1970's. The authors conclude that storage of high-level waste appear both technically and economically attractive. However, further investigation of methods to control radiolytic gas production and to minimize cavity alteration are required for liquid waste disposal. For solids, the chief remaining uncertainty is the combined effect of heat and overburden pressure on the structural stability of mined openings. (RAF)

Thermal Conductivity; Diffusivity; Temperature

WASTE STORAGE; SITE SELECTION; WASTES, HIGH-LEVEL; WASTES, SOLID; WASTES, LIQUID; SALT DEPOSITS; MINES; LABORATORY STUDIES; FIELD STUDIES; THERMAL PROPERTIES; CAVITIES; EVAPORATION; CONDENSATION; ADSORPTION; GASES; RADIATION DOSE; STRUCTURAL STABILITY; HEAT TRANSFER; CALCULATIONS; PLASTICITY; CHEMICAL PROPERTIES; RADIATION EFFECTS; PRESSURE EFFECTS; ECONOMICS; COSTS

<361>
Bradshaw, R.L., and W.C. McClain, Oak Ridge National Laboratory, Oak Ridge, TN.

Results and Analysis.

ORNL-4555 (Draft); ORNL-CF-70-7-42; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 11, (pp. 11.1-11.229), 822 pp.(1971, April)

Results and analysis of the experimental aspects of the project are covered in this chapter. Selected data and interpretations are presented to illustrate the behavior of the various phenomena under investigation, including the effects of radiation from the RTR fuel cycle, moisture, and temperature on rock salt. Rock deformation and mine stability results are described and discussed in detail, including a description of the various gauge types used, design of the gauge layout, and presentation of the data for all phases of the experiment. (CAJ)

Summary plots, tables, and graphs are included.

WASTES, RADIOACTIVE; WASTES, SOLID; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; MINES; DESIGN; MOISTURE; THERMAL PROPERTIES; CORROSION; DEFORMATION; ROCK MECHANICS; EQUIPMENT; RADIATION EFFECTS; INSTRUMENTS; THEORETICAL STUDIES

<362>
Bredehoeft, J.D., A.W. England, D.B. Stewart, N.J. Trask, and I.J. Winograd, U.S. Geological Survey.

Geologic Disposal of High-Level Radioactive Wastes - Earth-Science Perspectives.

U.S. Geological Survey Circular 779; 15 pp.(1978)

The difficulties and uncertainties connected with the geologic disposal of high-level radioactive waste are discussed. One major problem in this area is the credibility of geologic predictions over extended time spans. Past geologic events such as faulting, seismicity, or climate change probably have not been random, but deterministic explanations for their frequency, place of occurrence, magnitude, and rate of change are difficult to establish and these factors could drastically affect the geologic containment of high-level radioactive waste. Many questions concerning the behavior of rock salt must also be resolved. Rock salt has a high solubility and if relatively small amounts of brine can cause substantial decrease in the mechanical strength and possible movement of waste during a relatively short time, special efforts will surely be necessary to insure retrievability from a salt repository for periods as short as 10-25 years. The task of

<362>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<362> CONT.
characterizing ground water transport systems around potential repositories is also of the utmost importance. This task should be recognized as costly and time consuming, but empirical data on flow through fractured media and on the detailed chemistry of sorption-desorption phenomena for the actinide elements needs to be collected and used in conceptual models. In addition to these factors, more tools to evaluate potential repositories should be aggressively developed and more emphasis is needed on the relatively short and long term effects of the repository structure and the waste in the environment around the repository. Construction of a repository and emplacement of waste will initiate complex processes that cannot, at present, be predicted with certainty. The inability to predict can be offset in part by adoption of multiple-barrier or "defense in depth" philosophy. Such a philosophy provides a succession of independent barriers to radionuclide migration. The waste form, the host rock, and the groundwater flow path all provide potential barriers. (JST)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; GEOLOGY; HYDROLOGY; SALT DEPOSITS; REVIEWS; SOLUBILITY; CONTAINMENT; PREDICTIONS; REPOSITORY; HOST ROCKS; SITE SELECTION

<363>
Brewer, L.W., and J.H. Metcalf, Sandia Laboratories, Industrial Hygiene Division, Albuquerque, NM; Sandia Laboratories, Instrumentation, Dosimetry and NTS Projects Division, Albuquerque, NM.

Environmental Monitoring at the Waste Isolation Pilot Plant Site, Calendar Year 1976.

SAND-77-1021; 99 pp.(1977, August)

Measurements of environmental radiation levels at the WIPP site for CY 1976 are presented. Background radiation measured with the Reuter-Stokes Pressurized Ion Chamber is 6-20 uR/hr during the year, averaging 8.5 uR/hr (74 mR/yr). Quarterly measurement of 6 thermoluminescent dosimetry stations yielded values of 7.6-12.4 uR/hr (18,526.5 mR/yr); airborne gross beta was 0.016-0.427 pCi/cu m (monthly averages), with an average for the year of 0.02 pCi/cu m. There was an increase of 20-25% in TLD radiation levels in the fourth quarter, probably due to Chinese nuclear tests in September and November. Gross beta concentrations increased to a maximum of 1.07 pCi/cu m after the first test, and at the end of the year remained 3 times higher than levels before the tests. Average 24-hour concentrations of NO2, SO2, and H2S for the period sampled were 32.19 ug/cu m (maximum 130 ug/cu m), 8.29 ug/cu m (maximum 28 ug/cu m), and 0.110 ug/cu m (maximum 0.22 ug/cu m), respectively. Ozone concentrations were 0.013-0.032 ppm (monthly 24-hr average) for the period sampled. Atmospheric sampling was also carried out for CO and particulates. Estimated background annual whole-body radiation exposure to populations in the vicinity of the WIPP site is 66-84 mrad/yr from external sources. An additional 20 mrad/yr might be received via internal exposure (mainly to K 40). (LKH)

BACKGROUND RADIATION; EXPLOSIONS, NUCLEAR; ENVIRONMENT; OZONE; PARTICLES, AIRBORNE; SULFUR COMPOUNDS; NITROGEN COMPOUNDS; EFFLUENTS, AIRBORNE; DOSE RATE; MEASUREMENTS; EXPOSURE, INTERNAL; EXPOSURE, EXTERNAL; FIELD STUDIES; SITE SELECTION; WIPP; BETA PARTICLES

<364>
Briggs, L.I., Jr., University of Michigan, Ann Arbor, MI.

Geology of Subsurface Waste Disposal in Michigan Basin.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 128-153), 253 pp.(1968, August)

The suitability of an area for waste disposal must be based first on geologic factors such as: a) the presence of deep porous and permeable sandstones with sufficient fluid capacity and receptivity for long-term high-volume fluid disposal, b) the presence of thick salt deposits at a shallow depth for storage of stable solids in cavities, and c) the presence of thick shales for injection of grouted wastes into hydraulically produced fractures. In southeastern Michigan in the structural and sedimentary Michigan Basin in which strata dip less than 1 degree toward the basin center, though there are gentle open folds and few high-angle faults, the proper geologic conditions are present with a) the Mount Simon Member of the thick Cambrian sandstones, b) the salt deposits of the Silurian Salina Formation, and c) the Ordovician Utica Shale. The Cambrian sandstones are considered to be the best possibility for waste disposal, and most notably the Mount Simon Sandstone. It is a fairly consistent feldspathic, quartzose sandstone cemented largely by quartz and feldspar but locally by dolomite and calcite. Where it grades into carbonate facies it is less good as a reservoir. Glauconite is abundant in all of the Cambrian sandstones, and could be beneficial in the adsorption of radioactive cations from wastes. Although elsewhere in the state the formation is too deep for economic disposal of wastes and is used for potable fresh water, in southeast Michigan, in the Detroit area, it is 61 - 152 m (200 - 500 ft) thick, 762 - 2134 m (2500 - 7000 ft) deep, and is not a source of fresh water. In one core which included the Mount Simon, taken by the Consumers Power Company, porosity and permeability measurements were done. For the Cambrian sandstones the overall average porosity was 10.7%, and the overall average permeability was 32 md. Receptivity measurements showed an injection rate of 930 bbl/day at a surface pump pressure of 1000 psig. After acid treatment and hydraulic fracturing, receptivity increased to 1600 bbl/day at a surface pump pressure of 825 psig. More tests are needed for porosity, permeability, and receptivity. In addition to the disposal of fluid wastes into the Cambrian sandstones, the Utica Shale could take low-volume radioactive liquid wastes as grout injected into hydraulically produced fractures. The clay minerals would be beneficial in adsorption of radioactive cations. The Salina salt beds, largely halite, but containing layers of dolomite and anhydrite, could be used for storage of stable solidified wastes in cavities. (DCW)

This paper provides 28 maps, diagrams, and

<364> CONT.
tables showing geologic, hydrologic, mineralogic, petrologic, population distribution, and land use information.

Porosity, Permeability; Receptivity

WASTE STORAGE; WASTE DISPOSAL; WASTES, RADIOACTIVE; WASTES, LIQUID; WASTES, SOLID; STORAGE, GEOLOGIC; SALT DEPOSITS; CAVITIES; SEDIMENTARY ROCKS; SANDSTONES; SHALES; BASINS, STRUCTURAL; GEOLOGIC FORMATIONS; GEOLOGIC STRUCTURES; GEOLOGY; SITE SELECTION; POROSITY; PERMEABILITY; GROUTING; HYDRAULIC FRACTURING; GLAUCONITE; ABSORPTION; MEASUREMENTS; RECEPTIVITY; FRACTURES; INJECTION

<365>
Brown, P.M., D.L. Brown, N.S. Reid, and O.B. Lloyd, Jr., U.S. Geological Survey, Washington, DC.

Evaluation of the Geologic and Hydrologic Factors Related to the Waste Storage Potential of Mesozoic Aquifers in the Southern Part of the Atlantic Coastal Plain, South Carolina and Georgia.

U.S. Geological Survey Professional Paper 1088; 37 pp. (1979)

This report describes the subsurface distribution of rocks of Cretaceous to Late Jurassic (?) age in the Atlantic Coastal Plain of South Carolina and Georgia, and examines their potential for deep-well waste storage. For mapping purposes, a waste storage "operational unit" is established and defined. It is a sand or sandstone layer, 6.1 m (20 ft) or more in thickness, that is immediately underlain and overlain by a layer of shale or clay 6.1 m (20 ft) or more in thickness, and which occurs in regional chronostratigraphic units (Units A, B, C, D, E, F, G(?) and H(?)) of Mesozoic age in areas where each of these units contains nonusable ground water. Nonusable ground water is defined as water that contains NaCl in excess of 10,000 mg/l. Using a group of geohydrologic parameters derived from or combining 21 categories of basic data established from study and interpretation of well cuttings and geophysical logs, a series of 32 regional maps and 8 stratigraphic cross sections was constructed. For each of the eight geologic units delineated in the subsurface, the maps illustrate the distribution of waste storage potential in terms of areal extent, depth below land surface, sand-shale geometry, and the approximate NaCl concentration of a unit's nonusable ground water. In areas where the geologic units contain nonusable ground water, the depth below land surface and the thickness of potential waste-storage reservoir and reservoir seal combinations are variable. The range in variability appears to be broad enough to meet the need for a wide choice among the geologic requirements that would normally be considered in selecting specific waste storage sites for detailed examination. (Auth) (LKN)

Extensive well data are provided in tabular form.

Aquifer Thickness; Stratigraphic Unit Thickness

GEOLOGY; STRATIGRAPHY; LITHOLOGY; SEDIMENTARY ROCKS; GROUND WATER; BRINES; SANDS; SANDSTONES; SHALES; CLAYS; SEDIMENTS; MAPS; HYDROLOGY; GEOPHYSICAL SURVEYS; BOREHOLES; WELLS; GEOLOGIC MAPPING; GEOLOGIC STRATA; SITE SELECTION; WASTE DISPOSAL; WASTE STORAGE; WASTES, LIQUID; WELLS, INJECTION

<366>
Brown, P.M., and N.S. Reid, U.S. Geological Survey.

Geologic Evaluation of Waste-Storage Potential in Selected Segments of the Mesozoic Aquifer System Below the Zone of Fresh Water, Atlantic Coastal Plain, North Carolina through New Jersey.

Geological Survey Professional Paper No. 881; 47 pp. (1976)

Presented are the data from a reconnaissance survey conducted on the Atlantic Coastal Plain from North Carolina to New Jersey. The survey is a geologic evaluation of waste storage potential in portions of the Mesozoic aquifer system in the area. General geologic information needed for such an evaluation is: understanding of the geometry of the sediment mass; the conditions existing during its deposition; and how it was affected and modified by tectonic and structural events during and after its deposition. To delineate rocks high in permeability and porosity from rocks low in these qualities, knowledge is needed of their external and internal geometry and of their distribution within the sediment mass. Also the formation must be isolated hydrologically and must have a favorable ion exchange with the intended radwaste. In selecting the reservoir environment, the potential reservoir rock should be underlain and overlain by seal rocks. The seal rocks must be characterized by relatively low permeability, an absence of fractures, and sufficient thickness and lateral extent to qualify. Ideally, the reservoir rock should be connected laterally with rock having the capacity to accept the native fluid that moves out of reservoir storage in response to injection pressure. This will reduce the danger of fracturing and breaching of the reservoir seals. A preliminary definition of the required environment is a sand or sandstone layer 20 ft or more in thickness that is directly underlain and overlain by a shale or clay layer 20 ft or more in thickness. Units of interest in the study area are the Late Jurassic to Early Cretaceous periods. Since the fresh water environment must be protected, the fresh-saline water boundary zones within these three units was delineated. From control wells the following data were determined by geological and geophysical methods: unit thickness and sand thickness; average depth to the tops of the unit's potential waste-reservoir sand; thickness of potential waste-reservoir sand; thickness of overlying shale per foot of potential waste-reservoir sand; thickness of underlying shale per foot of potential waste-reservoir sand; overburden thickness per foot of potential waste-reservoir sand. This data is intended to aid management in deciding the favorable storage areas. This decision would be made within a cost-risk-benefit framework for detailed drilling of those areas most favorable. (SDV) (RAF)

Maps are included in back pocket of report.

Porosity; Stratigraphic Unit Thickness; Hydraulic Conductivity; Permeability; Depth to

<366> CONT.
Water Table

AQUIFERS, CONFINED; BEDROCK; CONTAINMENT; CONTAMINATION; COST BENEFIT ANALYSIS; DISPOSAL SITE; EVALUATION; FIELD STUDIES; FRESHWATER SYSTEMS; GEOLOGY; GEOPHYSICAL SURVEYS; HYDROLOGY; ION EXCHANGE; LITHOLOGY; OVERBURDEN; PERMEABILITY; POROSITY; SEDIMENTS; SITE SELECTION; WASTE DISPOSAL; WASTES, LIQUID; WASTES, RADIOACTIVE

<367>
Brown, R.J. (Ed.), National Technical Information Service, Springfield, VA.

Ocean Waste Disposal (A Bibliography with Abstracts).

NTIS/PS-77/0661; 297 pp. (1977, August)

This updated bibliography contains 297 abstracts, 66 of which are new entries to the previous edition. It was prepared by NTIS from its on-line interactive bibliographic retrieval system comprising more than 500,000 document/data records of federally sponsored research reports dating from 1964. Most of the cited topics in this bibliography involve ocean disposal of sewage, sewage sludge, and dredged material, although reports on the disposal of radioactive wastes, brines, and industrial wastes are also covered. Ecological effects are included, as is research on the pollution of New York Bight. Studies on the discharge of heated effluents are excluded. (Auth) (LEN)

Supersedes NTIS/PS-76/0558, NTIS/PS-75/462, and CCM-78-11265.

SEA DISPOSAL; BIBLIOGRAPHIES; WASTES, RADIOACTIVE; WASTES, NONRADIOACTIVE; WASTES, COMMERCIAL; WASTES, ORGANIC; WASTES, SOLID; WASTES, LIQUID; WASTE DISPOSAL; SLUDGES; BRINES; ECOSYSTEMS, AQUATIC; BIOSPHERE; POLLUTION, WATER

<368>
Brerton, G.D., and W.C. McClain, Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

Geological Criteria for Radioactive Waste Repositories.

Y/OWI/TM-47; 12 pp. (1977, November 28)

A comprehensive list of geological criteria and factors is provided so specifications can be derived in evaluating geologic formations as repositories. The geometry of the repository rock is described as that rock which is deep enough to separate it from surficial processes; which has sufficient vertical extent to preclude breaching of the geological containment during excavation or during the radioactive heat production; and which extends laterally far enough to provide adequate space to develop and operate the repository with a buffer zone of rock. The stability of the formation will be determined by the rate and amount of predictable regional uplift or subsidence of the bedrock; faults or other structural characteristics that will not compromise the repository operations; and igneous activity below that which would breach geologic containment. The hydrology of the repository will be such that ground water will not transport radionuclides to the biosphere and the water content of the repository rock will be low so underground chemical reactions will not occur as a result of radioactive heating. The rock in the repository will not react with the stored wastes or the canisters. Mechanical properties, state of stress, seismicity, and site development will also be taken into account. Socioeconomic criteria such a mineral resources, water resources, and land resources will be considered. (NEW) (BAF)

CONTAINMENT; CONTAINER INTEGRITY; DISPOSAL SITE; EXCAVATION; FAULTS; GEOLOGIC FORMATIONS; GEOLOGY; GROUND WATER; HEAT TRANSFER; HYDROLOGY; REPOSITORY; ROCKS; SEISMOLOGY; SITE EVALUATION; STORAGE, GEOLOGIC; WASTE STORAGE; WASTES, RADIOACTIVE; ECONOMICS

<369>
Bertholdez, B.C., J. Greenberg, J.A. Stottlemyre, D.J. Bradley, J.H. Raymond, and R.J. Serne, Battelle-Pacific Northwest Laboratories, Richland, WA.

Long-Term Safety Assessment of Geologic Isolation Repositories.

Transactions of the American Nuclear Society 28:83-84. (1978)

The Waste Isolation Safety Assessment Program (WISAP) has been initiated at Battelle-Pacific Northwest Laboratories to provide methods and data for making long-term safety evaluations of specific nuclear waste isolation systems. Major areas of research in this program are described. Development of a new technique called Repository Simulation Analysis, which attempts to combine the best features of expert opinion/Delphi, fault/event tree analysis is under way. A first-generation repository simulation computer code has been developed. An evaluation of computer codes available for estimating the consequences of hypothetical release scenarios (subsurface hydrologic, nuksurface contaminant transport, surface water dose, and air dose) has been completed, and ten codes have been selected for incorporation into an integrated modeling system. Development of a model to describe the accumulation and dispersion processes occurring when contaminated water is used for crop irrigation over long time periods is nearly complete. A team of geologic consultants is developing quantitative relationships between the potentially disruptive phenomena involved in simulation of the long-term behavior of geologic repositories. They are also developing the geologic and geophysical data needed to specify the parameters of the repository simulation model. Static, IAEA, and selected flow and anoxic leaching experiments for Tc and actinide-doped glasses have been started. Static and IAEA leaching experiments are under way for clad and unclad spent fuel. Similar studies will be performed for actual high-level waste glasses. Static, IAEA, and selected flow leaching experiments have been started for TRU waste forms of 4 types: concrete, bitumen, urea formaldehyde, and polymer. Results from these studies are too preliminary to be reported at present. Experimental methods for measuring the retention of radionuclides by geomedia are being investigated; preliminary results show that batch, low-pressure flow-through column,

<369> CONT.
and axial filtration methods yield nearly the same sorption values. The sorption of 15 radionuclides by more than 30 geologic materials, involving over 350 combinations of nuclides, geologic materials, and conditions, has been measured. A data bank is being established to organize these and future experimental results. (Auth)(LKM)

SAFETY; COMPUTER CODES; HAZARD ANALYSIS; MODELS; MODELS, MATHEMATICAL; RADIONUCLIDE MIGRATION; DISPERSION; CROPS; IRRIGATION; REPOSITORY; STORAGE, GEOLOGIC; LEACHING; GEOLOGY; SPENT FUELS; GLASS; CONCRETES; BITUMENS; UREA FORMALDEHYDE; POLYMERS; SORPTION; WASTE MANAGEMENT; LABORATORY STUDIES; THEORETICAL STUDIES; ACTINIDES; TECHNETIUM; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE

<370>
Byerly, D.W., Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

Report of Investigation Underground Limestone Mines in the Ohio Region.

Y/OWI/SUB-8251/2; 20 pp.(1976, June)

The geologic setting of several underground limestone mines in Ohio other than the PPG mine at Barberton was investigated. The sites visited included: 1) the Jonathan Mine located near Zanesville and currently operated by the Columbia Cement Corporation; 2) the abandoned Alpha Portland Cement Mine located near Ironton; and 3) the Lewisburg Mine at Lewisburg and currently utilized as a underground storage facility. Other remaining possibilities where limestone is being mined underground are located in middle Ordovician strata near Cornstown and Maysville, Kentucky. These are drift-mines into a thick sequence of carbonates. The author predicts that these mines would have source problems with water due to the preponderance of carbonate rocks and the proximity of the mines to the Ohio River. None of the sites visited nor the sites in Kentucky have conditions comparable to the deep mine at Barberton, Ohio. (Auth)(RAP)

LIMESTONES; MINES; GEOLOGY; MAPS; FIELD STUDIES

<371>
Byerly, D.W., Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

Cursory Search for Carbonates Suitable for Developing Dry Underground Storage Space in the Midcontinent, U.S.A..

Y/OWI/SUB-8251/3; 59 pp.(1975, November 15)

This search for carbonate strata for possible use as a nuclear waste repository was prompted by studies of two sites: a deep limestone mine near Barberton, Ohio, and near-surface drift limestone mines near Kansas City, Missouri. Geomorphology, bedrock geology, tectonics and seismicity, and hydrology of each site are reviewed. The two sites exhibit certain similarities that make them dry: geomorphological status, presence of impermeable shales, general structural setting, and low seismic risk. However, the Barberton site offers the greater potential for dry storage over a longer period of time. Both mine sites have impermeable shale above the mine, but the greater stratigraphic continuity and greater thickness of the shales at Barberton offers a much more certain safety barrier. Criteria developed for dry limestone excavations include: 1) presence of low-angle regional dips; 2) presence of thick sequences of impermeable shale or rocks that can serve as an impervious barrier above the excavations; 3) presence of glacial covers if high enough above the excavated horizon; 4) presence of plateau-like regions; 5) absence of complex folds and faults; 6) low seismic risk; 7) presence of carbonate rocks consisting of massive, dense limestone free from insoluble residue; 8) absence of jointing; and 9) avoidance of areas of considerable well drilling. (RAP)

Portions of tables and illustrations are illegible.

STORAGE, GEOLOGIC; LIMESTONES; REPOSITORY; GEOPHYSICAL SURVEYS; GEOLOGY; BEDROCK; TECTONICS; SEISMOLOGY; HYDROLOGY; SHALES; GEOLOGIC STRUCTURES; STRATIGRAPHY; GEOLOGIC MAPPING; FIELD STUDIES

<372>
Carruthers, J.

Nuclear Waste: Twenty-Five Years Hence ... And Beyond.

The Canadian federal and Ontario governments have agreed to initiation of studies by Atomic Energy of Canada, Ltd. (AECL) and Ontario Hydro on radwaste immobilization and geologic disposal techniques, and on interim storage and transportation of spent fuel. Over the next 3 years, AECL will do test drillings in syenites, anorthosites, gabbros, and granites in an effort to find 50-100 suitable plutons for geologic disposal sites. Information will be gathered on fracture density properties, chemical properties, groundwater migration, and geologic history. Heat distribution studies are planned using an INCO mine at Sudbury. Other scientists will be developing and testing designs for underground repositories. The most promising design currently under study would bury 1,082,000 canisters 0.3 m in diameter by 0.3 m high in holes drilled in the floors of rooms 1000 m underground; there would be 500 canisters/room, with 80 separate panels of 50 rooms each. An estimated 15 million cu m of rock would be excavated for the rooms, and another 500,000 cu m for the storage holes. Cost would be $400-800 million. Depending on the effectiveness of the bedrock as a barrier to migration, burial rooms might be filled with water-absorbing clays, or the waste containers enclosed in stainless steel (at an extra cost of $500 million). Since a decision has not been made about reprocessing of used CANDU reactor fuel, AECL is studying two options: 1) long-term retrievable storage of fuel rod assemblies after a 5-year temporary storage, and 2) burial of only the wastes left over from spent fuel reprocessing. Waste storage would be in the form of glass. AECL foresees the following timetable for the program: 1) concept verification (test drilling, pathway analysis, etc.) - 3 years; 2) site selection and confirmatory testing - 2 years; 3) design

<372> CONT.
and construction of a demonstration repository (holding the equivalent of 4 years' spent fuel from one Pickering-type CANDU reactor) and of a fuel and waste immobilization pilot plant (1/100 scale), at costs of $50-100 million and $15-20 million, respectively - 5 years; and finally, the commercial phase - establishment of large-scale commercial facilities. The entire program will require 25 years to achieve operable full-scale facilities. (LKH)

IMMOBILIZATION; STORAGE, GEOLOGIC; SPENT FUELS; TRANSPORTATION; DRILLING; SYENITES; ANORTHOSITES; GABBROS; GRANITES; IGNEOUS ROCKS; FRACTURES; CHEMICAL PROPERTIES; GROUND WATER; THERMAL PROPERTIES; DESIGN; REPOSITORY; CANISTERS; ECONOMICS; CONTAINERS; CONTAINMENT; RETRIEVABILITY; GLASS; WASTE MANAGEMENT; WASTE DISPOSAL

<373>
Carter, R.L., R.E. Goodson, and R.H. Herrill, State University of New York, Department of Earth and Space Sciences, Stony Brook, NY.

Summary Review of Rock Mechanics Workshop on Radioactive Waste Disposal.

V/ONWI/TM-20; CONF-761236; Rock Mechanics Review/Workshop, Denver, CO, December 16, 1976, (7 pp.). (1977, February 7)

A summary of work done on salt for radioactive storage is presented. For an experimental program uniaxial and triaxial compression of polycrystalline aggregates from Lyons, Kansas (bedded salt), Delaware Basin, New Mexico (bedded salt), and Jefferson Island, Louisiana (dome salt) were measured. Project Salt Vault was conducted in the Lyon salt; the Delaware Basin salt is a candidate for Sandia's WIPP project; and Jefferson Island material is being tested for comparisons of mechanical behavior of bedded and dome salts. Under the conditions employed (23 to 200 degrees C and atmospheric to 3000 psi), the salt specimens remained in the transient creep regime and steady-state creep was not observed. All samples had a very low yield strength and exhibited strain hardening to maximum compressive strain. Salt strength has a marked dependence on confining pressure or deviatoric loading path. All specimens deformed homogeneously. The modelling studies are at an advanced stage. In general, especially for the temperature and thermoelastic room deformation histories, agreement with Salt Vault results was excellent. An in situ experiment is planned for Avery Island, Louisiana dome salt mine to compare with and to extend results from Project Salt Vault in bedded salt. The test will be performed at the 500 ft level of a 50 yr-old mine. Halite (salt) is a reasonable choice for a repository by 1985 because it has a relatively high thermal conductivity, is relatively homogeneous with few discontinuities to allow migration of solutions, is relatively weak mechanically so large initial deviatoric stresses can not be maintained, and there are several existing mines in which prototype studies may be conducted. Recommendations by the committee are: establishment of a permanent advisory panel; complete cooperation between commercial radwaste disposal (ONWI) and military disposal (Sandia); removal of division between geology/hydrology and rock mechanics; and consideration of other materials such as shale, limestone, basalt, and granites. (NDV)

COMPUTER PROGRAMS; CONTAINMENT; DISPOSAL SITE; CREEP; GEOLOGY; MODELS, MATHEMATICAL; ROCK MECHANICS; SALT DEPOSITS; SALT DOMES; REPOSITORY; SHALES; LIMESTONES; BASALTS; GRANITES; STRAIN; STRENGTH, COMPRESSIVE; DEFORMATION; TEMPERATURE

<374>
Chan, T., N.G.W. Cook, and C.F. Tsang, Lawrence Berkeley Laboratory, Berkeley, CA.

Theoretical Temperature Fields for the Stripa Heater Project, Volume 1.

LBL-7082 1/2; SAC-09; 53 pg. (1978, September)

This report discusses thermal conduction calculations for three in-situ heater experiments being conducted at the Stripa Mine in Sweden. A semianalytic solution based on the Green's function method has been developed for an array of arbitrary time-dependent finite line heaters in a semi-infinite medium. This method as well as a three dimensional numerical model using the IFD (Integrated Finite Difference) technique has been applied to model the field situations at Stripa. Comparison has demonstrated that the finite line source solution for rock temperature is in excellent agreement with the numerical model solution as well as with a closed form finite cylinder source solution. It was found that maximum temperature rise in the rock within the two year experiment period will be 178 degrees C for the 3.6 kW full-scale heater experiment, 385 degrees for the full-scale experiment with a 5 kW central heater and eight 0.72 kW peripheral heaters, and less than 200 degrees C for the time-scaled experiment. The ring of eight peripheral heaters in the second full-scale experiment will provide a nominally uniform temperature rise within the perimeter a few weeks after turn-on. The high temperature zone is localized throughout the duration of all three experiments. Nevertheless, the effect of different spacings on the thermal interaction between adjacent radioactive waste canisters will be demonstrated by the time-scaled experiment. Detailed results are presented in the form of tables, temperature profiles and contour plots. Predicted temperatures have been stored in an on-site computer for real-time comparison with field data. (Auth) (LKH)

HEAT SOURCES; THEORETICAL STUDIES; MODELS; MODELS, MATHEMATICAL; THERMAL PROPERTIES; COMPUTERS; TEMPERATURE; SITE SELECTION; STORAGE, GEOLOGIC; ROCK MECHANICS; GRANITES; IGNEOUS ROCKS; REPOSITORY

<375>
Chapman, N., D. Gray, and J. Mather, Institute of Geological Sciences, London, England.

Nuclear Waste Disposal: The Geological Aspects.

New Scientist 78(1100):225-227; Atom 261:186-188. (1978, April 27; 1978, July)

This article outlines the objectives of high-level radioactive waste disposal, the

<375> CONT.

current policy to incorporate wastes into borosilicate glass blocks for pre-disposal surface storage to dissipate heat before permanent geologic burial, various disposal options, geologic criteria to be met by possible high-level waste disposal sites, the best kinds of candidate host rocks and a general discussion of where these are found in the United Kingdom, and finally the broad-based research and development program being funded by the EEC and the United Kingdom Atomic Energy Authority being carried out over the next 20 years by the Institute of Geological Sciences. The high-level wastes under consideration are by-products of reprocessed spent reactor fuel which contain fission products and actinides. The most feasible disposal option is deep geologic disposal in arrays of underground mined cavities 300 to 1000 m deep in rock suitable for emplacement of vitrified blocks. The geologic criteria to be met by a disposal site include proper volume and ease of mining access, lack of faults and fractures, no groundwater or surface water access, impermeable barriers, good thermal response of the rock, provision of a geochemical barrier to migration of actinides or fission products by means of ion-exchange or adsorption, lack of waste/rock chemical interactions that would jeopardize the integrity of the rock, chemical stability of the rock over time, low seismicity, low commercial significance in terms of mining or other interests, and ability to withstand major long-term climatic changes. The research and development program is aimed primarily at examining the suitability of crystalline igneous or metamorphic rocks for waste disposal, with a secondary look at argillaceous rocks and salt deposits. With laboratory, theoretical, and field studies, the group hopes to learn specific details about the behavior of rocks under high-level waste repository conditions. Factors to be investigated include: groundwater; fluid/rock and fluid/waste interactions; thermal stress, convection, and permeability modelling; thermal stability and thermomechanical properties of the rock; backfilling and sealing methods; three-dimensional borehole mapping techniques; rock-heating; and use of microseismic monitoring systems. Detailed investigations would be carried out on specific sites. (DCW)

This article was first published in New Scientist and was reprinted in Atom with the only change being the removal of a list of the geological characteristics of a possible high-level waste disposal site.

WASTE STORAGE; WASTE DISPOSAL; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; GEOLOGY; ROCKS; GLASS; SHALES; SALT DEPOSITS; IGNEOUS ROCKS; METAMORPHIC ROCKS; SEDIMENTARY ROCKS; HOST ROCKS; METHODS; SITE SELECTION; LABORATORY STUDIES; THEORETICAL STUDIES; FIELD STUDIES; CAVITIES; THERMAL PROPERTIES; GROUND WATER; SPENT FUELS; REPOSITORY; FISSION PRODUCTS; ACTINIDES; ABSORPTION; WASTE-ROCK INTERACTIONS; MONITORING; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; GEOCHEMISTRY

<376>

Cherry, J.A., R.W. Gillham, G.E. Grisak, and E.L. Lesh, University of Waterloo, Waterloo, Ontario; Alberta Environment, Lethbridge, Alberta; Beak Consultants Limited, Mississauga, Ontario.

A Concept for Long-Term Isolation of Solid Radioactive Waste in Fine-Grained Deposits.

Management of Low-Level Radioactive Wastes, R.B. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 10, (pp. 1021-1040), 1214 pp.

As an alternative to the 'bury above the water-table' approach, it is proposed that solid low-level radioactive wastes be buried at the bottom of large holes augered to depths of many meters below the water-table in fine-grained deposits. In southern Canada the best deposits in this regard are dense relatively unfractured clayey till and glaciolacustrine clay [hydraulic conductivity 10(E-8) - 10(E-10) cm/sec; hydraulic gradient 0.1-1; porosity 0.1-0.2%]. Below the zone of water-table fluctuation the mechanism of radionuclide transport will be molecular diffusion. Diffusion is a process that is so slow that radionuclides such as Sr 90, Cs 137, and H 3 will decay to levels that are very near background before migration to the biosphere. Long lived isotopes such as C 14 and Ra 226 will eventually diffuse to the biosphere, but only after many thousands of years. The emission flux will be very low. In hydrogeologic studies of prospective sites for below-water-table disposal in clayey deposits, determination of the age of the groundwater by means of the naturally occurring isotopes H 3, O 18, H 2, and C 14 is a key factor. The burial zones must be shown to have pore water that is very old, thereby indicating extremely small groundwater velocities (on the order of 0.2 - 0.02 cm/yr). (Auth) (LKH)

BURIAL, INTERMEDIATE; SATURATED ZONE; CLAYS; TILL; SEDIMENTS; DIFFUSION; RADIONUCLIDE MIGRATION; GROUND WATER; AGE DETERMINATION; HYDRAULIC VELOCITY; HYDRAULIC GRADIENT; POROSITY; HYDRAULIC CONDUCTIVITY; THEORETICAL STUDIES; EQUATIONS; RECOMMENDATIONS; WASTE DISPOSAL; WASTE MANAGEMENT

<377>

Claassen, H.C., U.S. Geological Survey.

Hydrologic Processes and Radionuclide Distribution in a Cavity and Chimney Produced by the Cannikin Nuclear Explosion, Amchitka Island, Alaska.

U.S. Geological Survey Professional Paper 712-D; 28 pp. (1978)

An analysis of hydraulic, chemical, and radiochemical data obtained in the vicinity of the site of a nuclear explosion (code-named Cannikin, 1971), on Amchitka Island, Alaska, was undertaken to describe the hydrologic processes associated with the saturation of subsurface void space produced by the explosion. Immediately after detonation of the explosive, a subsurface cavity was created surrounding the explosion point. This cavity soon was partly filled by collapse of overburden, producing void volume in a rubble chimney extending to land surface and forming a surface-collapse sink. Surface and ground water immediately began filling the chimney but was excluded for a time from the cavity by the presence of steam. When the steam condensed, the accumulated water in

<377>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<377> CONT.
the chimney flowed into the cavity region, picking up and depositing radioactive materials along its path. Refilling of the chimney voids then resumed and was nearly complete about 260 days after the explosion. The hydraulic properties of identified aquifers intersecting the chimney were used with estimates of surface water inflow, chimney dimensions, and the measured water-level rise in the chimney to estimate the distribution of explosion-created porosity in the chimney, which ranged from about 10 percent near the bottom to 4 percent near the top. Chemical and radiochemical analyses of water from the cavity resulted in identification of three aqueous phases--ground water, surface water, and condensed steam. Although most water samples represented mixtures of these phases, they contained radioactivity representative of all radioactivity produced by the explosion. Sorption of radioactivity on particulate matter was evident, more than 10,000 times as much activity being found on solids as in solution; however, no selectivity for specific isotopes was observed and sorption equilibrium had not been reached 800 days after the explosion. Although large amounts of heat were released by the Cannikin explosion, approximately 90 percent of this heat was absorbed by rock and water that fell into the cavity. As heat in the cavity dissipated to the surroundings, additional voids were created by cooling of the rocks that had fallen into the cavity; this new space in part accounted for the continued water flow into the cavity observed after chimney filling was virtually complete. The occurrence of zones of low hydraulic conductivity may also have contributed to the delay in filling the cavity. (Auth)

Transmissivity; Porosity; Distribution Coefficient; Temperature

H 3; Cs 137; U

HYDRAULICS; RADIOCHEMISTRY; EXPLOSIONS, NUCLEAR; HYDROLOGY; CAVITIES; SURFACE WATERS; GROUND WATER; STREAM; RADIONUCLIDES; AQUIFERS; DISTRIBUTION; POROSITY; CHEMICAL ANALYSIS; SORPTION; PARTICLES; SOLUBILITY; HEAT SOURCES; ROCKS; HYDRAULIC CONDUCTIVITY; LITHIUM; SILICA; MAGNESIUM; CALCIUM; SODIUM; POTASSIUM; SULFATES; CHLORIDES; ALPHA PARTICLES; URANIUM; BETA PARTICLES; RADIATION, GAMMA; TRITIUM; CESIUM 137; IONS

<378>
Cobbs Engineering, Tulsa, OK.

Survey of Active and Inactive Mines for Possible Use as In Situ Test Facilities.

Y/OWI/SUB-76/16518; 140 pp. (1976, September)

As part of the National Waste Terminal Storage Program OCC-ND's Office of Waste Isolation subcontracted Cobbs Engineering to carry out a survey of active and inactive mines for possible use for in situ testing of geologic storage and retrievability of radioactive waste (including spent fuel, high-level waste, etc.). This preliminary report covers available information gathered from the literature, U.S. Bureau of Mines, Mining Enforcement and Safety Agency, and a limited number of personal contacts with constructors or operators of facilities. Included are descriptions of 288 facilities indexed by depth of mine, nature of the country rock, mineral mined, and type of entry. Mine summary data include: location (state and county), owner, country rock type, depth of mine, entries (type and number), mining plan, dates of first and last (if inactive) mining, volume of mine, mineral mined, present service, and a brief general description of the mine. The criteria for judging the suitability of a mine for experimentation are that the mine be dry, over 300 ft in depth, and its mineralization such that there is little, if any, prospect of contact between the experimental equipment and circulating ground waters. Results generally indicate sedimentary rock mines of minerals of evaporite origin as preferable for in situ testing, followed by other sedimentary rocks, and then by mines producing minerals from any rock type where the mine is above local water table. Fourteen inactive and 34 active mines have been selected as most worthy of further investigation for possible use as nuclear waste isolation test facilities. (Auth) (EKW)

STORAGE, GEOLOGIC; MINES; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; SPENT FUELS; RETRIEVABILITY; GEOLOGY; SEDIMENTARY ROCKS; EVAPORITES; GROUND WATER; SALT DEPOSITS; WASTE STORAGE; REVIEWS; SITE SELECTION

<379>
Colton, G.W., U.S. Geological Survey.

Geologic Summary of the Appalachian Basin with Reference to the Subsurface Disposal of Radioactive Waste Solutions.

TEI-791; 121 pp. (1961, June)

This report is an extensive review of the stratigraphy/sedimentology and structure of the Appalachian basin with a view toward identification of areas potentially suitable for geologic disposal of radioactive wastes. The waste disposal possibilities of the stratified rocks in the basin are considered in terms of the following: 1) gross lithology of the sequences; 2) general lithology of the rock units composing the sequences; and 3) the structural attitude of the sequences in different parts of the basin. The degree of exploitation of economically significant mineral resources is considered briefly where such exploitation may affect waste-disposal possibilities. Hydrologic aspects in general are not considered. Based largely on the above geologic factors the following types of reservoirs associated with particular geologic environments offer some prospects for the disposal of radioactive waste solutions: 1) artificially created cavities in thick salt beds; 2) artificially fractured thin lenticular sandstone bodies isolated in shale or mudstone sequences; 3) portions of thick noncarbonate clastic sequences possessing appreciable natural porosity and permeability; and 4) thin clastic units (with natural or artificially created openings) in the lower plate of a thrust fault overlain by impermeable strata. Considered in its entirety, the Late Ordovician clastic sequence appears to have a greater number of favorable geologic factors for waste disposal purposes than the others. This sequence consists largely of shale, siltstone, and sandstone; coarse-grained light gray to red rocks are common in the sequence along the eastern side of the basin, whereas

<379> CONT.
fine-grained dark to black calcareous rocks are common along the west side. The most favorable location for waste disposal in this sequence appears to lie in the region just west of the Catskill Mountains in south central New York and north central Pennsylvania (Queenston Shale, Oswego Sandstone, Reedsville Shale, and Utica Shale formations). Areas of possible local suitability occur along the axis of the Alleghany Mountains in eastern West Virginia and central Pennsylvania. The Early Silurian clastic sequence, the Silurian-Devonian Carbonate sequence, and the Devonian clastic sequence offer fewer possibilities. The Late Precambrian stratified sequence, Early Cambrian, and the Cambrian-Ordovician carbonate sequence offer few possibilities. The Mississippian and Pennsylvanian sequences appear to be generally unsuitable. (Auth) (LBH)

Several geologic maps and cross sections are provided in a pocket at the end of this report.

STRATIGRAPHY; SEDIMENTS; GEOLOGY; GEOLOGIC STRUCTURES; BASINS, STRUCTURAL; LITHOLOGY; ORE DEPOSITS; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; WASTE STORAGE; WASTE DISPOSAL; SITE SELECTION; SALT DEPOSITS; SANDSTONES; FRACTURES; CAVITIES; SHALES; FAULTS; SEDIMENTARY ROCKS; REVIEWS

<380>
Cook, N.G.W., Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA; University of California, Department of Materials Science and Mineral Engineering, Berkeley, CA.

An Appraisal of Hard Rock for Potential Underground Repositories of Radioactive Wastes.

LBL-7006; 15 pp. (1977, October)

The mechanical safety and stability of underground radioactive waste repository depends largely on the virgin state of stress in the rock, groundwater pressures, the strengths of the rocks, heating by the decay of the radioactive wastes, and the layout of the excavations and the disposition of waste canisters within them. Each of these factors has been analyzed in the light of data available in the literature. Virgin states of stress have been identified which would retard groundwater ingress and obviate the occurrence of faulting, even in the presence of water pressure equal to the hydrostatic head, in rock around repositories 0.5-2 km below surface. To meet these requirements the value of the minimum horizontal component of this state of stress must be greater than 2/3 that of the vertical component, and the maximum stress difference should be less than 25 MPa. It is recommended that the area of excavations in any one horizon occupy no more than 20% of the area of that horizon, to avoid adverse interaction between excavations. For depths up to 1.2 km, the laboratory uniaxial compressive strength of the rock should be 200 MPa. Neither local, short-term, nor regional long-term thermomechanical stresses appear to pose serious difficulties, provided power densities are kept below about 16 W/sq m and temperatures below 100 C. A single repository comprising 3 horizons each 5 sq km in area separated by about 200 m would have the capacity to store wastes with thermal output around 240 MW. The results indicate no fundamental geologic or mechanical reasons why waste repositories should not be excavated at suitable sites in hard rock. However, specific tests to determine the mechanical and thermal properties of the rocks at a site would be needed to provide data for the engineering design of a repository. Also, little experience exists of the effects on underground excavations of thermal loads, so that this aspect requires theoretical study and experimental validation. (Auth) (LBH)

SITE SELECTION; REPOSITORY; STORAGE, GEOLOGIC; ROCK MECHANICS; STRESS; IGNEOUS ROCKS; METAMORPHIC ROCKS; THERMAL PROPERTIES; PHYSICAL PROPERTIES; HYDROLOGY; FAULTS; HYDRAULIC HEAD; STRENGTH, COMPRESSIVE; THEORETICAL STUDIES; REVIEWS

<381>
Cook, N.G.W., and R. Wood, Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA; University of California, Department of Materials Science and Mineral Engineering, Berkeley, CA.

Full-Scale and Time-Scale Heating Experiments at Stripa: Preliminary Results.

LBL-7072; SAC-11; 13 pp. (1978, December)

Two full-scale heating experiments and a time-scale heating experiment have been started in drifts excavated in granite 340 m below the surface in the abandoned mine at Stripa, Sweden. The full-scale tests comprise electrical heaters with a maximum power output of 20 kW, to simulate high-level waste canisters 0.3 m in diameter by 2.5 m long (heated length). These heaters have been installed in vertical boreholes 0.4 m in diameter by 5.5 m deep drilled in the floor of the full-scale test drift, and separated by a distance of 22 m. Using the quadratic relationship between length and time in solutions to the linear conduction of heat, the linear scale of the time-scale heaters has been reduced by 0.31 of the full-scale, so that time is accelerated by a factor of 10.2. The purpose of the full-scale heating experiments is to assess the near-field effects of thermal loading for the design of an underground repository. The purpose of the time-scale experiments is to obtain field data on the interaction between heaters and their effect on the rock mass during a period of about 2 yrs, corresponding to about 20 yrs of full-scale operation. Preliminary results show that measured temperatures are slightly lower than predicted, due to water in the heater holes, causing the heat to be dissipated over a greater length of hole by evaporation at the heater and condensation further up the hole. The temperature measurements suggest that heat flow in the rock mass is predominantly by conduction and that theoretical values of thermal coefficients can be used to predict rock temperatures reasonably well. Measurements of rock displacement between anchor points 3 m below the heater mid-plane and the hole collar show values 1/2 to 1/3 of predicted values. A possible explanation for this might be thermal expansion of the rock mass into open joints. (LBH)

Temperature

HEAT SOURCES; HEAT TRANSFER; THERMAL PROPERTIES;

<381>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<381> CONT.
GRANITES; IGNEOUS ROCKS; TEMPERATURE; MINES; BOREHOLES; SITE SELECTION; STORAGE, GEOLOGIC; REPOSITORY; DESIGN; THERMAL CONDUCTIVITY; THERMAL EXPANSION; JOINTS; FIELD STUDIES

<382>
Cook, N.G.W., and P.A. Witherspoon, Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA; University of California, Department of Materials Science and Mineral Engineering, Berkeley, CA.

In Situ Heating Experiments in Hard Rock: Their Objectives and Design.

LBL-7073; SAC-10; Mechanical and Thermal Design Considerations for Radioactive Waste Repositories in Hard Rock, (pp. 17-26), 26 pp.(1978, October)

Of the many alternatives that are being considered for the disposal of nuclear wastes, deep underground burial is favored. The wealth of experience concerning the design and construction of underground excavations does not include the unique effects of heating excavations by radioactive decay, nor the issue of long-term isolation. The effects of heating are important in establishing the feasibility of this method of disposal, and are essential for the design of an underground repository. Near-field phenomena around individual canisters can be studied by full-scale experiments, using electrical heaters. The thermal diffusivity of rock is so low that information concerning the interaction between full-scale heaters and the effects of heating a large volume of rock cannot be measured in full-scale experiments lasting less than a few decades. To overcome this difficulty, a time-scaled heating experiment has been developed in which a reduction in linear scale is accompanied by an acceleration of the time scale to the second power. In this experiment, the linear scale is about a third, so that the time scale is about ten fold. (Auth)

ROCKS; HEAT SOURCES; EXCAVATION; STORAGE, GEOLOGIC; WASTE DISPOSAL; REPOSITORY; CONTAINERS; INSTRUMENTS; THERMAL PROPERTIES; DIFFUSION; TIME FACTOR; STRESS; THEORETICAL STUDIES; FIELD STUDIES; SITE SELECTION

<383>
Dames and Moore, Cincinnati, OH.

BWTS In-Situ Test Needs Workshop, August 29-30, 1978.

ONWI-7; 142 pp.(1978, December)

A workshop to identify major technical concerns in nuclear waste isolation that need to be resolved by in situ testing was held in Columbus, Ohio, August 29-30, 1978, under the sponsorship of the Battelle Memorial Institute Office of Nuclear Waste Isolation. Presentations were given on the status of in situ testing programs, including Project Salt Vault, WIPP, Avery Island, Hanford basalt tests, Stripa granite tests, Climax Granite tests, and Eleana/Conasauga Shale tests. Working groups were set up to develop prioritized lists of technical issues that need to be examined by in situ field testing. These groups were as follows: Salt - Heat Transfer and Thermal Analysis, Rock Mechanics, Near-Field Interactions, and Far-Field Interactions; Hard Rock - Thermal Analysis and Rock Mechanics and Waste/Rock Interactions; Argillaceous Rock - Thermal Analysis and Rock Mechanics and Waste/Rock Interactions. The workshop agreed that a definition of the source term - including waste form (age, chemical composition, isotopic composition, and packaging), canister material, and overpack and/or backfill material - was lacking. Also, it was felt that chemical and mechanical behavior in the near-field environment is not well understood and that in situ testing and demonstration projects would be necessary. In the far-field, the primary issue of concern was the characterization of the hydrogeologic environment, because of the need to assess nuclide migration phenomena. It was concluded that because of the time frame under consideration and the scale of the repository, specific in situ testing should be directed toward computer model verification and evaluation of scaling problems. It was also suggested that evaluation of time and geometric in situ phenomena occurring naturally should be pursued in order to better assess scale and time factors. (LPE)

Presentations for invited speakers on ongoing and planned in situ testing are included as an appendix.

WASTE STORAGE; STORAGE, GEOLOGIC; WASTE DISPOSAL; SALT DEPOSITS; WIPP; BASALTS; GRANITES; SHALES; HEAT TRANSFER; ROCK MECHANICS; WASTE-ROCK INTERACTIONS; CHEMICAL COMPOSITION; ISOTOPES; PACKAGING; CONTAINERS; OVERPACKING; BACKFILLING; HYDROLOGY; RADIONUCLIDE MIGRATION; COMPUTERS; MODELS, MATHEMATICAL; MODELS; TIME FACTOR; RECOMMENDATIONS

<384>
Dames and Moore, White Plains, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 8. Baseline Rock Properties: Salt.

E/GWI/GS-36/8; 128 pp.(1978, April)

Salt as a medium for a repository for storage of radwaste is investigated. Salt formations considered in this report are the Salina Basin Salt, Paradox Basin Salt, and a salt dome. Parameters measured, calculated and discussed are: mineralogy, lithology, major structures, weathering/alteration, hydrology/water content, jointing, unit weight, natural moisture content, porosity, Young's Modulus, Poisson's Ratio, bulk modulus, shear modulus, cohesion friction angle in the 1500-2500 psi range, uniaxial compressive strength, tensile strength, plastic behavior, coefficient of linear thermal expansion, heat capacity, thermal conductivity, horizontal permeability, and vertical permeability. The Saline Basin salt deposits are found in Michigan, Ohio, New York, Pennsylvania, and Ontario, Canada. Lithologically it is composed of halite, carbonates, and shales in a number of stratigraphic units. The salt deposits of the Permian Basin are considered from the sub-basin in Kansas and New Mexico. These deposits are the Hutchinson Salt Member of

<384> CONT.
the Wellington Formation and it consists of anhydrite beds alternating with gray and red shales. The primary unit of consideration in the Paradox Basin of Colorado and Utah is the Paradox Member of the Pennsylvanian Hermosa Formation. Lithologically the Paradox Member is a cyclothem of 29 cycles of anhydrite, dolomite and shale. The salt dome discussed is the Lomana underlying southern Arkansas, Mississippi, Alabama, Texas and Louisiana. A summary of special salt characteristics is provided. (NDV)

Extensive tabular data and technical figures are presented.

Thermal Conductivity; Thermal Expansion; Heat Capacity; Tensile Strength; Compressive Strength; Poisson's Ratio; Young's Modulus; Porosity; Volumetric Water Content

ANHYDRITE; CLAYS; DESIGN; ELASTOPLASTICITY; FIELD STUDIES; GEOLOGY; HYDROLOGY; THERMAL PROPERTIES; MOISTURE CONTENT; PHYSICAL PROPERTIES; PERMEABILITY; POROSITY; REPOSITORY; ROCK MECHANICS; SALT DOMES; SALT DEPOSITS; STRENGTH, COMPRESSIVE; STRENGTH, TENSILE; REVIEWS; HALITE; ISOTROPY; SITE SELECTION; SEDIMENTARY ROCKS; ROCKS; STORAGE, GEOLOGIC; ROCK MECHANICS

<385>
Dames and Moore, White Plains, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Stratigraphics of Salt, Granite, Shale, and Basalt.

T/OWI/TM-36/3; 65 pp.(1978, April)

The object of this study is to provide generic stratigraphic sections for a variety of geologic media (salt, granite, shale and basalt) to be used in preconceptual design studies of deep geologic disposal of radioactive waste. The term generic is intended to imply a lack of specificity; however, in some cases the siting criteria (depth, thickness, and lateral extent) conflict with the effort to maintain a lack of specificity, and the resulting generic stratigraphic sections are recognizable and reasonable only a limited number of geologic formations and geographic areas. The methodology used to develop the generic stratigraphic section consisted of a literature review from which two to six real stratigraphic sections were obtained for each area of interest. The real stratigraphic sections were compared and combined into a composite section generally representative of each study area. By comparing and combining the composite stratigraphic section in a subjective manner, the generic stratigraphic section was developed. Salt sections were taken from the Salina, Paradox, Permian, and Williston Basins. Argillaceous sections were taken from the Mid-continent area, Mississippi Embayment, Pierre Shale, Eocene Formation, and the Great Valley of California. The generic granite section was derived from investigations in the Maritimes, Appalachians, Great Lakes Shield area, Rockies, Sierra Nevada, and the Basin and Range Province. The basalt section was based on the Columbia River Group. (Auth)(LKB)

REVIEWS; SALT DEPOSITS; BASALTS; GRANITES; SHALES; IGNEOUS ROCKS; SEDIMENTARY ROCKS; EVAPORITES; STRATIGRAPHY; REPOSITORY; BASINS, STRUCTURAL; SITE SELECTION; STORAGE, GEOLOGIC

<386>
Dawson, F.M., and J.R. Tillerson, Sandia Laboratories, Albuquerque, NM.

Salt Motion Following Nuclear Waste Disposal.

CONF-780136; SAND-77-1226C; Evaluation and Prediction of Subsidence, Proceedings of an International Conference, Pensacola, FL, January 15-20, 1978, (20 pp.).(1978)

Two finite-element models have been applied to the analysis of bedded salt behavior in the presence of heat-generating radioactive wastes. One model uses a thermomechanical secondary creep approximation; behavior is represented by a nonlinear relationship between applied stresses and resulting strain rates. The flow fields and temperature distributions are coupled through temperature-dependent body forces, temperature-dependent constitutive equations, viscous dissipation, and changes in the system geometry; elastic response is neglected. The other model uses a thermoelastic formulation. Stresses and deformations are determined for elastic thermal strains due to changes in temperature; temperature distributions for times during the deformational history of the region are computed prior to the thermoelastic analyses and used within the formulation to evaluate thermal strains. The models have been applied to studies of storage alternatives and to analysis of buoyant movement of heated salt. The finite element models have been applied to the analysis of three problems associated with ground movement in a salt repository: an alternative study for canister emplacement options in which the temperature increases due to heat producing wastes accelerate creep deformations of different nine rooms; the buoyant effects produced by a single canister surrounded by salt; and the effects produced by an entire repository with a waste canister horizon. Results of the first study indicate that storing wastes horizontally in the pillar is less suitable than storage beneath the room. Wastes stored in the pillar increase creep rates, accelerating room closure. Also, maintaining a low extraction ratio instead of dispersing the canisters (high extraction ratio) allows longer times for retrievability. Storing wastes beneath the rooms while maintaining a large pillar width provides the best storage alternative of the cases considered. Transient analyses of individual canister movement show that the canister initially sinks with increasing velocity, until heating of the salt forms a convective cell, and the canister rises; eventually, heat production falls, and the canister sinks again. Total vertical displacement is less than 1 m. Steady-state analyses also indicate displacement is not significant. Buoyant motion on a regional scale for an axisymmetric region with a heat-producing horizon at 800 m depth and extending radially from 590 m to 1600 m from the shaft centerline was analyzed using both models. Initial thermal load is 25 W/sq m, with a 30 yr half-life. Using the thermomechanical secondary creep model, displacement after 400 yrs is on the order of 0.5 m, with or without a stiff layer overlying the salt bed. Thermoelastic

<386> CONT.
analyses show maximum displacement to be about 0.8 m. After the initial period of heating, creep strain increments are much larger than the elastic strain increments, and predominate through most of the deformational period. (Auth) (LKH)

See also Nuclear Waste Canister Thermally Induced Motion, P.R. Dawson and J.R. Tillerson, SAND-78-0566, June 1978.

MODELS; MODELS, MATHEMATICAL; STORAGE, GEOLOGIC; SALT DEPOSITS; WASTES, HIGH-LEVEL; HEAT TRANSFER; HEAT SOURCES; CREEP; THEORETICAL STUDIES; VISCOSITY; THERMAL PROPERTIES; WASTE STORAGE

<387>
Deju, R.A., Rockwell International, Atomics International Division, Richland, WA.

Basalt Waste Isolation Program Monthly Report, February 1978.

(RHO-BWI-78-100 FEB; 23 pp. (1978, February 28)

This report describes progress on the Basalt Waste Isolation Program at Hanford for the month of February, 1978. Twenty sq mi of the Saddle Mts. were geologically mapped at a scale of 1:24000; indications of Pliocene uplift leave the tectonic stability of the mountains an unresolved question. A preliminary characterization of 26 gravel pits in and near Hanford was made; no tectonic structures were found. Approximately 1800 wells which penetrate basalt in the Pasco Basin have been identified. Work is in progress on identification of wells drilled below the Saddle Mts. Formation-Wanapum Basalt contact. In the Waste/Basalt interaction study, work continued toward isolation of interaction products which are primary hosts for various ions. The following observations have been made: Ba is tied up in Ba-weeksite and Ba-pollucite; Cs occurs both in solution and in Ba-weeksite; B is tied up in tincalconite, and Fe in acmite. Mixtures of $SrZrO_3$ (believed to be the most likely Sr phase in unreprocessed fuel) and plagioclase have been treated hydrothermally at 200 and 300 C and 300 bars to explore whether Sr can be mobilized and if it will be fixed by ion exchange with plagioclase in basalt. Results after 7 days at 300 C and 300 bars indicate a decrease in X-ray crystallinity, but no obvious interaction. Hydrologic Hole DC-5 was completed at 3990 ft. An artesian zone above the Umtanum Unit was exposed to a lost circulation zone below the Unit just before the well was completed; an expandable rubber packer was run and set in the middle of the Umtanum Unit to prevent communication between zones. Borehole DC-6 is presently cored to 2583 ft; planned total depth is 4800 ft. Mine modeling studies have shown that, at depths of 3000 ft, probability of joint slip above the plane of the repository is small (safety factor = 2). A request for proposals to perform a "Disruptive Event Analysis" for a geologic repository has been prepared, as the first phase of a risk analysis which will include "transport modeling" and "consequence analysis" phases. Design of heaters and instrumentation for the Near Surface Test Facility continues. Procurement specifications are being prepared. A Tectonics and Seismicity of the Columbia Plateau Workshop was held in Seattle, WA, February 14-16, 1978. (Auth) (LKH)

FIELD STUDIES; LABORATORY STUDIES; GEOLOGIC STRUCTURES; WEEKSITE; BASINS, ARTESIAN; MODELS; THEORETICAL STUDIES; STRONTIUM; GEOLOGIC MAPPING; GEOLOGY; HYDROLOGY; CORES; FRACTURES; BASALTS; PHYSICAL PROPERTIES; THERMAL PROPERTIES; HYDROTHERMAL ALTERATIONS; FIXATION; GROUND WATER; MINERALS; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; AQUIFERS; DESIGN; STRATIGRAPHY; CHEMICAL ANALYSIS; BOREHOLES; GEOPHYSICAL SURVEYS; REPOSITORY

<388>
Deju, R.A., Rockwell International, Atomics International Division, Richland, WA.

Basalt Waste Isolation Program Monthly Report, January 1978.

RHO-BWI-78-100 JAN; 23 pp. (1978, January 31)

This report describes programs on the Basalt Waste Isolation Program at Hanford for the month of January, 1978. A revision of stratigraphic nomenclature for the Pasco Basin was completed, work was begun on geologic mapping of the Rattlesnake Hills area, and approximately 40 sq mi were geologically mapped at a scale of 1:24000 in the vicinity of Hanford. Several faults and folds were identified and mapped in the Nine Canyon area, the Saddle Mts., Umtanum Ridge, and Yakima Ridge. Fracture characteristics have been logged in 100 ft of DB-5 core, interflow units measured in outcrops of Grande Ronde Basalt at Sentinel Gap, gravel pits on the reservation have been characterized, and three 1000 ft magnetic profiles have been completed across the Zintel Canyon Fault south of Hanford. Chemical analyses have been performed on 58 basalt samples. In the hydrologic program, testing procedures were established and preparations begun to develop the tools needed for field testing. Drilling operations continue at Hydrologic Hole DC-5, which is at a depth of 3500 ft (goal is 4000 ft); cable tool operations at DC-8 have reached 428 ft (goal is 700 ft). Core Hole DC-11 at Gable Mountain was completed at 385 ft; Core Hole DC-6 has casing cemented to 1100 ft and drilling is at 1200 ft. Waste/basalt interaction studies continued, with hydrothermal treatment of simulated waste glass spheroids to determine reaction rate, solid phase alteration products, and the species mobilized into solution after 7 days at 300 C/300 bars. Testing to 14 days almost totally altered and fragmented the glass; weeksite crystals were present on the fragments and in the capsule. Cs mobilization was about 2% in 7 days and 12% in 14 days; no Sr was detected in solution. CsOH and Cs_2MoO_4 have been reacted with labradorite for 10 days at 200 C/300 bars, in both cases yielding pollucite $(Cs,Na)AlSi_2O_6$ as the prime reaction product; with Cs_2MoO_4 as the Cs-source, scheelite $CaMoO_4$ also formed. Large-volume reconnaissance experiments have been initiated. Development of computer codes to study the thermal and mechanical rock mass properties pertinent to the design and construction of a repository continues. A contract for the physical and thermal testing of rock cores was awarded to the Colorado School of Mines - Earth Excavation Institute. Core characterization is now complete and testing under way. A bid package for the construction of the Near

<388> CONT.
Surface Test Facility was prepared during the reporting period. (Auth) (LKM)

FIELD STUDIES; LABORATORY STUDIES; GRAVELS; CHEMICAL ANALYSIS; GLASS; CESIUM; STRONTIUM; PCLLUCTYR; SCHEELITE; COMPUTER CODES; GEOLOGIC MAPPING; GEOLOGY; HYDROLOGY; CORES; FRACTURES; BASALTS; PHYSICAL PROPERTIES; THERMAL PROPERTIES; HYDROTHERMAL ALTERATIONS; FIXATION; GROUND WATER; MINERALS; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; GEOLOGIC STRUCTURES; AQUIFERS; DESIGN; STRATIGRAPHY; BOREHOLES; GEOPHYSICAL SURVEYS; REPOSITORY

<389>
Deju, R.A., Rockwell International, Atomics International Division, Richland, WA.

Basalt Waste Isolation Program Monthly Report, March 1978.

RHO-BWI-78-100 MAR; 26 pp. (1978, March 31)

Approximately 15 square miles were geologically mapped during the reporting period in the Pasco Basin and vicinity. X-ray analysis and fracture mapping continued on schedule. Subsurface geological maps have been prepared for three stratigraphic horizons beneath the Pasco Basin. Preliminary hydrologic testing of a 600-foot interval at Core Hole DC-6 was conducted. Permeability measurements were generally below detection limits. At the end of the month, Core Hole DC-4 was at a depth of 3,000 feet, Core Hole DC-6 was at 3,183 feet, and Core Hole DC-8 was at a depth of 1,530 feet. The mechanical and thermal testing of cores from the Near-Surface Test Facility was completed. The results indicate that the Pomona flow has average basalt properties which vary little with location in the flow. Hydrothermal runs of basalt/spent nonprocessed fuel interactions and basalt/glass interactions were continued. Characterizations of the solid and products and the resulting solutions are under way. Design of heaters for the Near-Surface Test Facility is approximately 90 percent complete. Preliminary instrument borehole arrays for the full- and time-scaled tests have been defined. A preliminary computer analysis of the temperatures, stresses, and displacements induced in the rock mass by the full- and time-scaled heater tests has been completed. The design of Near-Surface Test Facility tunnels and excavations has been completed. Included in this design are approximately 5,000 lineal feet of holes drilled in the test rooms as part of the test program. Numerical modeling of rock stresses within a basalt waste repository was continued using finite element, finite difference, and displacement discontinuity techniques. Several two- and three-dimensional single-room and regional computer runs were performed. (Auth)

Illustrations show location of drill holes and a conceptual sketch of combined near-surface and spent fuel test facility.

Thermal Conductivity; Density; Porosity; Compressive Strength; Tensile Strength

BASALTS; GEOLOGIC MAPPING; FRACTURES; FIELD STUDIES; HYDROLOGY; PERMEABILITY; DRILLING; THERMAL PROPERTIES; SPENT FUELS; GLASS; BOREHOLES; STRESS; EXCAVATION; MODELS, MATHEMATICAL; REPOSITORY; INSTRUMENTS; HEAT SOURCES; SITE SELECTION; LABORATORY STUDIES; SEDIMENTS; GEOPHYSICAL SURVEYS; GEOLOGY; CORES; PHYSICAL PROPERTIES; STRENGTH, MECHANICAL; HYDROTHERMAL ALTERATIONS; FIXATION; GROUND WATER; WASTE DISPOSAL; WASTE STORAGE; DESIGN

<390>
Deju, R.A., Rockwell International, Atomics International Division, Richland, WA.

Basalt Waste Isolation Program Monthly Report, April 1978.

RHO-BWI-78-100 APR; 28 pp. (1978, April 30)

Mapping of Late Cenozoic sediments and underlying basalt continued. Field work is well underway. Specific geologic maps covering 100 square miles in the Wahluke Ridge/Saddle Mountain area of the Western Pasco Basin were completed. Data from wells penetrating basalt were organized for use in the three-dimensional modeling of the Columbia Plateau hydrology. Drilling operations were conducted in three locations. Cable tool operations at Well DC-4 progressed from 300 feet to 544 feet with a planned total depth of 600 feet. Coring advanced from 3,183 feet to 3,916 feet in Well DC-6 and from 1,530 feet to 1,612 feet in well DC-8. The planned depth at Wells DC-6 and DC-8 are 4,300 and 4,100 feet, respectively. A testing report on the mechanical and thermal properties of the Pomona basalt flow was completed. A program for borehole plugging for Hanford was written. This program is specific to the physical and chemical conditions found in boreholes penetrating the basalt and sedimentary rock sequence at Hanford. Reconnaissance hydrothermal experiments have been completed utilizing waste/basalt mixtures. The analysis of these runs is continuing with particular attention being given to recognizing trends within given pairs of waste/basalt mixtures as a function temperature. A series of long-duration experiments with waste/basalt pairs has also been initiated. Some high-temperature equilibrium experiments using basalt and a simulated ground water composition have been completed. Design of the full- and time-scale heater tests for the Near-Surface Test Facility was also completed. Final design revisions were accomplished on the tunnels and excavations document for the Near-Surface Test Facility. Site-survey staking was completed and the approved design of the electric distribution line was released as ready for construction. Effort on preparation of function and design criteria for the repository was continued. When complete, the document will contain a statement of the functions of the repository and design criteria suitable for concept design considerations. (Auth) (RAP)

The location of drillholes is illustrated.

Compressive Strength

SEDIMENTS; SEDIMENTARY ROCKS; BASALTS; FIELD STUDIES; WELLS; FRACTURES; MODELS; HYDROLOGY; DRILLING; THERMAL PROPERTIES; GROUND WATER; EXCAVATION; REPOSITORY; INSTRUMENTS; HEAT SOURCES; SITE SELECTION; GEOLOGIC MAPPING; LABORATORY STUDIES; GEOLOGY; CORES; PHYSICAL PROPERTIES; STRENGTH, MECHANICAL; HYDROTHERMAL ALTERATIONS; FIXATION; MINERALS; WASTE DISPOSAL; WASTE STORAGE; DESIGN; BOREHOLES; GEOPHYSICAL

<390>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<390> CONT.
SURVEYS

<391>
DeLaguna, W., Oak Ridge National Laboratory, Health Physics Division, Oak Ridge, TN.

Importance of Deep Permeable Disposal Formations in Location of a Large Nuclear-Fuel Reprocessing Plant.

American Association of Petroleum Geologists, Memoir 10: Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 21-31), 253 pp. (1968, August)

The most important problem in location of a large nuclear-fuel reprocessing plant is that of low-level radioactive waste, which comes from three general sources: condensate containing large quantities of tritium and some fission products from the evaporation process used to reduce the volume of high-level waste, contaminated water containing variable quantities of fission products, and chemical waste solutions. Since large quantities of water can dilute these wastes, disposal into large bodies of surface water has been common; however, underground disposal into deep permeable formations seems to offer the best possibility for disposal of low-level wastes. Suitable deep permeable formations for waste injection through wells could be in structural or sedimentary basins in which the geologic structure and hydrology are fairly simple and there would be a high degree of containment. Additional geologic information on porosity would be needed, but advantages for low-level waste disposal would be the lack of surface water contamination, the ability to provide less expensive containment of a major accident, and the opportunity to locate a plant away from large bodies of surface water. Other disposal methods for low-level wastes have been a) treatment for removal of strontium and cesium and discharge into a river, b) discharge into cribs and seepage pits where soils and clays absorb radionuclides, c) shallow land disposal, and d) discharge into the ocean. For disposal of intermediate-level wastes the most favorable method is injection of waste/cement slurries into hydraulically fractured shale. A favored method for disposal of high-level wastes is storage in solid form in mined salt cavities, or alternatively in mined rock cavities. Since the high-level wastes can be transported, they are not critical to site selection for a reprocessing plant. Maximum permissible concentrations of radioactive nuclides in air and water have been derived for people employed in the nuclear industry, but as better information becomes available, these values may change and may alter requirements for a plant site. Public attitudes about safety of the populace are also factors in site selection. Disposal of radioactive wastes into deep permeable formations is now practicable only for low-level wastes, but it holds potential for disposal of certain intermediate and high-level wastes, as well as for such gaseous wastes as krypton 85. (Auth) (DCW)

This paper provides three tables: one each to list compositions of typical low-level, intermediate-level, and high-level radioactive wastes. Seven references are given.

WASTE DISPOSAL; WASTE STORAGE; WASTES, RADIOACTIVE; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, SOLID; WASTES, GASEOUS; SITE SELECTION; STORAGE, GEOLOGIC; GEOLOGIC STRUCTURES; BASINS, STRUCTURAL; HYDRAULIC FRACTURING; SLURRIES; SHALES; SEDIMENTARY ROCKS; WELLS, INJECTION; POROSITY; SALT DEPOSITS; CAVITIES; ROCKS; METHODS; CRIBS; SEEPAGE PITS; BURIAL, SHALLOW; SURFACE WATERS; SAFETY; SOCIAL ASPECTS; CESIUM; KRYPTON; STRONTIUM; TRITIUM; GEOLOGY; WASTE TREATMENT; GROUTING; PERMEABILITY; SURFACE WATERS; INJECTION

<392>
Dolgov, V.V., B.S. Kolychev, V.V. Konstantinovich, V.V. Kulichenko, B.V. Nikipelov, A.S. Nikiforov, V.P. Martynov, S.N. Ozirsner, V.M. Sedov, and V.G. Shatsillo, Union of Soviet Socialist Republics, State Committee for Utilization of Atomic Energy, Moscow, USSR.

Development of Solidification and Burial Methods for Radioactive Wastes of Fuel Cycle.

IAEA-CN-36/350; Nuclear Power and its Fuel Cycle, Proceedings of an International Conference, Salzburg, Austria, May 2-13, 1977. International Atomic Energy Agency, Vienna, (19 pp.) (1977)

The USSR is developing processes of production of phosphate and silicate glassy materials for use in the isolation and final disposal of high- and intermediate-level radioactive wastes. The two-stage process of waste incorporation into glass is developed with the use of fluidized-bed calcination at the first stage followed by calcinate melting in a concrete crucible. For the purpose of reliable burial of intermediate-level liquid wastes, a solidification process has been developed for incorporation of the waste into bitumen, instead of storing the waste as liquid in stainless steel tanks. In addition to the development of glass incorporation methods for high-level liquid wastes, methods of safe and reliable storage and burial are also being studied. The storage is located on the ground grade, above the water table. As heat release rates decrease, the concrete compartments of the storage facility filled with the glassy wastes are transferred to facilities for permanent storage. These permanent storage facilities have provisions for retrieval built in. Studies of final disposal of radioactive bitumens were also conducted. It was found that disposal of the bitumens in clayey soil was feasible. (JMT)

BITUMENS; CALCINATION; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTE STORAGE; WASTE DISPOSAL; SOLIDIFICATION; BURIAL; GLASS; RETRIEVABILITY; CLAYS; WASTES, LIQUID

<393>
Droste, J.B., University of Indiana, Bloomington, IN.

Paleozoic Stratigraphy of Two Areas in Southwestern Indiana.

Y/OWI/SUB-706212; 53 pp. (1976, September)

Two areas recommended for evaluation as solid

<393> CONT.
waste disposal sites lie along the strike of Paleozoic rocks in southwestern Indiana. The two areas are designated northern and southern in this report. Crane Naval Ammunition Depot in Martin County, Indiana is the northern area. The three separate but nearby sites in Crawford and Perry Counties, Indiana are combined into the southern area. Thin Pennsylvanian rocks and rocks of the upper Mississippian are at the bedrock surface in extremely dissected uplands in both areas. The gross subsurface stratigraphy beneath both areas is the same, but facies and thickness variation in some of the subsurface Paleozoic units provide for some minor differences between the areas. Thick Mississippian carbonates grade downward into clastics of lower Mississippian (Borden Group) and upper Devonian (New Albany Shale) rocks. Middle Devonian and Silurian rocks are dominated by carbonate lithologies. Upper Ordovician (Maquoketa Group) overlie carbonates of Middle Ordovician age. Both areas will provide two rock zones with suitable geological and engineering properties to permit development of repository facilities for underground waste disposal. The major advantage of the northern area is that the land is now federally owned. (WAF)

Maps are given showing the thickness distribution and structures of geologic groups found in both northern and southern areas.

Stratigraphic Unit Thickness

STRATIGRAPHY; WASTES, SOLID; WASTES, RADIOACTIVE; REPOSITORY; SITE SELECTION; GEOLOGY; WASTE DISPOSAL; BEDROCK; SHALES; SILTSTONES; SANDSTONES; LIMESTONES; GEOLOGIC STRUCTURES; OUTCROPS; GEOLOGIC HISTORY; WELLS; FIELD STUDIES

<394>
Edmund, R.W., and E.D. Goebel, Augustana College, Rock Island, IL; Kansas State Geological Survey, Lawrence, KS.

Subsurface Waste-Disposal Potential in Salina Basin of Kansas.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 154-164), 253 pp. (1968, August)

A very general overview of the geology of the Salina Basin in Kansas shows good potential storage reservoirs for industrial waste in the Cambrian Reagan Sandstone, in Pennsylvanian shale beds, and in the Late Permian Hutchinson Salt. The Reagan Sandstone may constitute a common reservoir with the underlying pre-Upper Cambrian metasediments of the Rice Formation and the overlying Late Cambrian - Early Ordovician Arbuckle Dolomite. The Reagan and Arbuckle indicate good transmissibility characteristics from brine-disposal histories, and are overlain by the impermeable Middle Ordovician Simpson Group shales. The Reagan varies from a few feet of detrital sand to a mere quartzite 30 m (100 ft) thick to a detrital arkosic sandstone grading into metasediments. The Middle Pennsylvanian shale beds seem suitable for injection of liquid industrial waste slurries into artificial fractures. The shale is 30 - 91 m (100 - 300 ft) thick, and 914 m (3000 ft) deep. The Hutchinson Salt seems suitable for storage of solid or liquid industrial wastes. It has a maximum thickness of 91 m (300 ft) and is within 305 m (1000 ft) of the surface in the southwestern part of the basin. Not many data are available on fluid flow characteristics in the basin, but the natural flow is thought to be from north to south at a slow rate. (Auth) (DCW)

This paper does not specifically mention radioactive wastes, but the geologic conditions discussed could apply to radioactive wastes. Six maps and one geologic cross section are given.

WASTE STORAGE; WASTE DISPOSAL; WASTES, LIQUID; WASTES, SOLID; WASTES, COMMERCIAL; SALT DEPOSITS; SEDIMENTARY ROCKS; SANDSTONES; SHALES; DOLOMITE; BASINS, STRUCTURAL; GEOLOGIC FORMATIONS; GEOLOGY; SLURRIES; FRACTURES; GROUTING; INJECTION

<395>
Empson, F.B., and W.C. McClain, Oak Ridge National Laboratory, Oak Ridge, TN.

Mine Renovation and Preparation of Experimental Area.

ORNL-4555 (Draft); ORNL-CP-70-7-42; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 6, (pp. 6.1-6.29), 822 pp. (1971, April)

Presented in this chapter is the work required to prepare the Lyons Mine as a repository. Renovation was necessary on the head frame, hoist, and electrical system. A waste-charging shaft was drilled with no more than 1 degree of variance from the vertical. Cleaning and restoring the mine level entries and access ways, power and ventilation were also done. Once the restoration was finished the excavation and preparation of the experimental area was begun, and progress was noted. During this time 20,000 tons of salt were removed from the experimental area. (NDV)

Several photos, figures, and tables are included.

WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; MINES; MINING

<396>
Fenix and Scisson, Incorporated, Mercury, NV.

Minability Study of the Terminal Waste Storage Program, Eleana Investigation.

NVO-0038-35; 47 pp. (1978, April)

A minability study was undertaken to determine if shafts, drifts and large open chambers could be mined in the Eleana argillite, and to estimate mining costs. The assumption was made that the facility would contain 30,000 canisters emplaced at the rate of 1,000/yr in holes drilled both horizontally in the pillars and vertically down in the floor of the emplacement rooms at a 5 m spacing. In area 3,500 ft by 10,000 ft (800 acres) would be required. Not included

<396> CONT.
in the study were the effects due to seismic activity or thermal loadings. Based on present, incomplete information it is concluded that much of the ground appears to be poor due to extensive faulting and fracturing; tunneling will be slow, difficult and expensive. The total estimated cost for disposal of 30,000 waste canisters using the conceptual configuration assumed for this study, is $811,350,000. Time required for initial development work prior to first canister emplacement is estimated to be 2.22 years. (RAP)

Drawings are given of conceptual layout, shaft area and panel development, typical storage rooms and general ventilation flow.

MINING; COST BENEFIT ANALYSIS; CONTAINERS; DRILLING; CORES; FAULTS; FRACTURES; TIME FACTOR; ROCKS; EXCAVATION; ARGILLITES; QUARTZ; STRESS; GEOLOGIC MAPPING

<397>
Fritz, P., J.F. Barker, and J.E. Gale, University of Waterloo, Department of Earth Sciences, Waterloo, Ontario, Canada.

Geochemistry and Isotope Hydrology of Groundwaters in the Stripa Granite: Results and Preliminary Interpretation.

LBL-8285; 135 pp.(1979, April)

This report is a first summary of hydrogeochemical and hydrogeologic investigations at the Stripa mines, Sweden, during FY 1978. Data obtained from these investigations of the granitic rock mass at Stripa should provide the basic framework for geochemical studies needed in projects predicting containment movements from disposal sites in similar rock masses. Groundwater samples were collected from shallow, private wells, surface boreholes, and boreholes drilled from the 330 m and 410 m levels in the mine. Analyses included: field measurements of alkalinity, Eh, pH, conductance, temperature, and dissolved oxygen; complete major ion analyses; determination of composition of dissolved gases on a selected number of samples, with emphasis on noble gas analyses; H 2 and O 18 analyses; analyses of C 13 in dissolved aqueous carbon; and analyses of C 13 and O 18 from fracture calcites. In addition, attempts were made to determine groundwater ages through analyses of H 3, C 14, U isotopes, and U daughter products. Total dissolved solids in all groundwaters analyzed to date is below 500 mg/l, with chemical characteristics that change from calcium bicarbonate waters near the surface to sodium-calcium-chloride waters at depth. The principal change observed in the chemical load with depth was chloride concentration, which increases from less than 5 mg/l to about 300 mg/l. There is a parallel increase in pH, from about 6.5 to over 9.75. This contrasts with the much lower pH values observed in most other crystalline terrains, and will have important implications for the design of containment systems. Calcite saturation is maintained but, because of rising pH, dissolved organic carbon is lost. The total carbonate content thus decreases from about 70 mg/l to less than 7 mg/l. Oxygen 18 and H 2 analyses demonstrate that different fracture systems contain different water masses, whose ages increase with depth. Groundwater age determinations with C 14 and isotopes of the U decay series indicate that water ages exceed 25,000 years. The C 13 contents of the aqueous carbonate in these groundwaters indicate recharge through vegetated soil, presumably during an interglacial period. Carbon 13 and O 18 determinations show that most of the fracture calcites have formed in a wide variety of depositional environments, and not in the waters circulating today; some appear to have precipitated from subglacially recharged waters, while others show evidence of strongly reducing conditions. (Auth) (LKM)

Total Ion Concentration; pH; Calcium Carbonate Content; Total Dissolved Solids

HYDROLOGY; GEOCHEMISTRY; GRANITES; IGNEOUS ROCKS; GROUND WATER; ALKALINITY; pH; Eh; TEMPERATURE; DEUTERIUM; OXIGEN 18; CARBON 13; CARBON; CALCITE; CARBON 14; TRITIUM; URANIUM; DAUGHTER PRODUCTS; AGE DETERMINATION; CHLORIDES; CONCENTRATION; TOTAL ION CONCENTRATION; CALCIUM CARBONATE CONTENT; FRACTURES; RECHARGE; TOTAL DISSOLVED SOLIDS; WASTE STORAGE; SITE SELECTION; STORAGE, GEOLOGIC; WELLS; BOREHOLES; FIELD STUDIES; LABORATORY STUDIES

<398>
Gale, J.E., and P.A. Witherspoon, University of Waterloo, Department of Earth Sciences, Waterloo, Ontario, Canada; Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA.

An Approach to the Fracture Hydrology at Stripa: Preliminary Results.

LBL-7079; SAC-15; 26 pp.(1979, May)

There are two main problems associated with the concept of geologic storage of radioactive waste in fractured crystalline rock: (1) the thermo-mechanical effects of the heat generated by the waste, and (2) the potential for transport of radioactive materials by the groundwater system. In both problems, fractures play a dominant role. An assessment of the hydraulic and mechanical characteristics of fractured rock requires a careful series of laboratory and field investigations. The complexity of the problem is illustrated by the field studies in a fractured granite that are currently underway in an abandoned iron-ore mine at Stripa, Sweden. Such information is being gathered from an extensive series of boreholes and fracture maps. The approach being taken to integrate these data into an analysis of the fracture hydrology is reviewed and preliminary results from the hydrology program are presented. (Auth)

SITE SELECTION; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; FRACTURES; THERMAL PROPERTIES; FIELD STUDIES; ROCKS; GRANITES; HYDRAULICS; BOREHOLES; DRILLING; EQUATIONS; PERMEABILITY; VENTILATION; GROUND WATER; ROCK MECHANICS; GEOLOGIC HISTORY; TRITIUM; OXYGEN 18; OXYGEN 16; CARBON 14; CARBON 13; URANIUM 234; URANIUM 238; HELIUM; ARGON

<399>
Galley, J.E.

Economic and Industrial Potential of Geologic

<399> CONT.
Basins and Reservoir Strata.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 1-10), 253 pp.(1968, August)

This paper is introductory to the American Association of Petroleum Geologists Memoir 10, and discusses one potential value of subsurface strata in geologic basins as being the available space for storage or disposal of industrial waste liquids and solids (including radioactive wastes). Disposal methods which are being developed include deep-well injection into permeable formations, storage of solids in salt caverns, and pumping of liquid waste/cement slurries into artificially produced fractures in shale where they harden. Six of the papers in the memoir describe geologic basins where such disposal methods might be successful. Physical limitations to these methods include restricted space capacities in reservoir strata, pressure limits, possibilities of escape through pressure-induced fractures, adverse hydrodynamic conditions, incompatibility of injected fluids with reservoir fluids, and plugging of rock pores by precipitates or other materials. These limitations and the principles governing subsurface-disposal techniques are discussed in more detail in the three papers following this in the memoir. Several attributes of geologic basins over other structures for waste disposal include fewer faults, fewer unconformities and stratigraphic discontinuities or truncations, and fewer abrupt or less numerous lateral transitions in lithofacies. These conditions are relevant to the uniqueness of the hydrologic system. There is an important need for subsurface geologic investigations in the process of selecting sites for facilities that produce toxic wastes. (Auth) (DCW)

WASTE DISPOSAL; WASTE STORAGE; WASTES, COMMERCIAL; WASTES, LIQUID; WASTES, RADIOACTIVE; WASTES, SOLID; WELLS, INJECTION; GEOLOGIC FORMATIONS; GEOLOGIC STRUCTURES; BASINS, STRUCTURAL; GEOLOGY; ROCKS; METHODS; FIELD STUDIES;THEORETICAL STUDIES; SITE SELECTION; HYDRODYNAMICS; HYDROLOGY; SHALES; SALT DEPOSITS; SANDSTONES; FAULTS; HYDRAULIC FRACTURING; SLURRIES; SEDIMENTARY ROCKS; FRACTURES; GROUTING; INJECTION

<400>
Galley, J.E. (Ed.), American Association of Petroleum Geologists, Tulsa, OK.

Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata.

American Association of Petroleum Geologists, Memoir 10; 253 pp.(1968, August)

This collection of ten papers is a result of a request in 1958 by an agency of the U.S. Atomic Energy Commission for studies of geologic basins in which safe underground disposal of radioactive waste might be attempted on an experimental basis. The investigation was to employ modern methods of subsurface analysis developed by petroleum geologists. Because the techniques proposed for disposal of radioactive waste are generally applicable to many other industrial wastes, the emphasis in these papers is on subsurface disposal of industrial wastes in general. Six of the papers are studies of geologic basins in the U.S. The remaining four discuss principles involved in deep geologic disposal of wastes, such as the economic and industrial potential of geologic basins and reservoir strata, subsurface disposal of liquid industrial wastes by deep-well injection, the importance of deep permeable disposal formations in location of a large nuclear-fuel reprocessing plant, and fluid mechanics of deep-well disposal. Careful selection of disposal sites and reservoir strata, and selection of sites for the plants which produce toxic wastes, based on thorough subsurface geologic investigations, may help to lessen industry's problems of pollution of the surface environment. Nine papers were separately abstracted and entered into the data base. (Auth) (DCW)

Though the papers on Pennsylvania, the Salina Basin, the Denver Basin, and the San Juan Basin do not specifically mention radioactive waste, the principles and geologic conditions also apply to radioactive waste. The one paper not separately abstracted was on fluid mechanics of deep-well disposal.

Porosity; Permeability; Receptivity

WASTE DISPOSAL; WASTES, COMMERCIAL; WASTES, RADIOACTIVE; WASTES, LIQUID; GEOLOGIC STRUCTURES; SITE SELECTION; WELLS, INJECTION; NUCLEAR FACILITIES; ROCKS; GEOLOGY; THEORETICAL STUDIES; METHODS; FIELD STUDIES; BASINS, STRUCTURAL; SEDIMENTARY ROCKS; STORAGE, GEOLOGIC; GEOLOGIC FORMATIONS; PERMEABILITY; POROSITY; WASTE STORAGE; INJECTION

<401>
Garbarini, G.S., and H.K. Weal, Sun Oil Company, Denver, CO; Wolf Exploration Company, Dallas, TX.

Potential of Denver Basin for Disposal of Liquid Wastes.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 165-185), 253 pp.(1968, August)

A reconnaissance subsurface geologic study has shown that three types of reservoirs are available for liquid-waste disposal in the Denver Basin: fractured Precambrian rocks, porous sandstones, and thick shale suitable for disposal by the hydraulic-fracturing technique. From early 1962 through early 1966, fractured Precambrian rocks at a depth of 3658 m (12000 ft) were used as a disposal reservoir for toxic effluents produced at the Rocky Mountain Arsenal near Denver. The disposal well is now shut pending investigation of the possible relationship of waste injection to Denver-area earthquakes, which increased in frequency and magnitude during the injection period. Two borings on the Apishapa uplift indicate a good fractured Precambrian reservoir, though data are sparse. Porous sandstone reservoirs considered most favorable for waste disposal are the Permian Lyons Sandstone, the Triassic Dockum sandstone, the Triassic-Jurassic Jelm-Entrada sandstone, Cretaceous Dakota

<801> CONT.
Group sandstones, and the Hygiene zone of the Pierre Shale. The Lyons, Dockum, and Dakota are best suited for waste disposal in the southern part of the basin because of the relative lack of oil and gas exploration. The Dockum Sandstone, potentially the best disposal reservoir volumetrically, is limited to the southeast part of the basin. The Jelm-Entrada and Hygiene zone sandstones are potential disposal reservoirs along the heavily populated strip between Denver and Cheyenne. Cretaceous marine black shale suitable for disposal by the hydraulic-fracturing and grout-injection technique is present everywhere in the basin. The shale crops out over large areas, but beneath the populous strip along the Front Range, the shale is covered locally by as much as 610 m (2000 ft) of Upper Cretaceous and Tertiary strata. (An editorial note indicates that since 1966 oil and gas exploration has increased markedly along the Las Animas arch in the southeastern part of Colorado, statements in the text about the low level of activity in this area are no longer true.) (Auth) (DCW)

Though radioactive wastes are not specifically mentioned in the paper, the geologic conditions discussed could also apply to radioactive wastes. The paper provides 47 references and 12 illustrations, including 3 cross sections, to show the general geology of the Denver Basin.

WASTE DISPOSAL; WASTE STORAGE; WASTES, LIQUID; BASINS, STRUCTURAL; GEOLOGIC FORMATIONS; GEOLOGIC STRUCTURES; GEOLOGY; ROCKS; SEDIMENTARY ROCKS; SHALES; SANDSTONES; GROUTING; HYDRAULIC FRACTURING; SITE SELECTION; FRACTURES; EARTHQUAKES; WELLS, INJECTION; INJECTION; POROSITY; STORAGE, GEOLOGIC

<802>
Gera, F., CNEN-CSN Casaccia, Rome, Italy.

Radioactive Waste Disposal in Geological Formations.

IAEA-CN-36/313; Nuclear Power and Its Fuel Cycle, Proceedings of an International Conference, Salzburg, Austria, May 2-13, 1977. International Atomic Energy Agency, Vienna, (12 pp.). (1977)

The major source of radwastes that need to be disposed of are produced by the electro-nuclear industry. High-level wastes pose the most difficult technical problem. These wastes contain more than 99.9% non-volatile fission products, about 0.5% of uranium and plutonium, and almost all the other actinides formed in the fuel by transmutation of uranium and plutonium. From inhalation hazard potential and ingestion potential hazard computations, disposal of hundreds of cubic meters of short-lived wastes should be of a few hundred years. The few tens of cubic meters of long-lived wastes should be confined about a hundred thousand years. The short lived wastes could be disposed of by: shallow land disposal in trenches or concrete pits; sea dumping; or emplacement in deep cavities free from circulating ground water. Several ideas have been proposed to dispose of long-lived wastes, such as geologic disposal and extraterrestial. At present the geological disposal concepts are feasible. They are: disposal into geological formation underlying the sea bed; disposal in glaciated areas; and disposal into geological formations underlying the land areas. For sea bed disposal the areas would have to be in mid-gyre and mid-plate for the most stable regions. However, there are severe philosophical problems with using the ocean floor for dumping. While ice sheets are unlikely to be used by man, limited understanding of the geophysics of large ice-caps and the mechanisms controlling the long term climate eliminate this as a present possibility. Also, an international agreement forbids dumping of radwastes in Antarctica. Rock salt, argillaceous sediments and hard rocks (igneous and metamorphic) are the prime formations to be used for disposal under land area. Rock salt is usually completely free of circulating ground water and its plasticity seals fractures and openings. However, salt is highly soluble and other potential uses may be more important. Argillaceous sediments have pore water migration which over geologic time is significant but sorption by the clay minerals would reduce the radwaste migration rate. These sediments are also self-healing. The disadvantages to using argillaceous sediments are mining problems and less effective containment of certain radionuclides. Undoubtedly there are important questions about the effects of heating and radiation on the rock and the pore fluids which must be resolved by research and development. Hard rocks have fracture problems and no plasticity. Also there are questions about the possible effects of heat expansion and its consequences for the sealed fractures. In doing a safety analysis it is hard to predict the climate changes and whether or not future man will intrude. (SDV)

ACTINIDES; BEDROCK; BURIAL; CONTAINMENT; DEMONSTRATION; DISPOSAL SITE; ENVIRONMENT; FISSION PRODUCTS; GEOLOGY; GROUND WATER; HAZARD ANALYSIS; HYDROLOGY; IGNEOUS ROCKS; METAMORPHIC ROCKS; RADIONUCLIDE MIGRATION; SAFETY; SALT DEPOSITS; SEAWATER; SHALES; WASTE DISPOSAL; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, SOLID; REVIEWS; SITE SELECTION

<803>
Gera, F., G. Lenzi, L. Sensi, and G. Cassano, CNEN-CSN Casaccia, Rome, Italy; CNEN-CSN Trisaia, Matera, Italy.

Disposal of Long-Lived Radioactive Wastes in Italy.

CONF-760310; STI/PUB/431; IAEA-SM-207/48; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 2, (pp. 259-273), 428 pp. (1976)

The Trisaia Center in southern Italy is the site of CNEN's research and development program for geologic disposal of long-lived radioactive wastes. The site is in an area which is tectonically stable (Mercalli Scale 7), and is located near the Ionian Sea on a quaternary terrace which is underlain by 800-1000 m of Plio-Pleistocene early clay. Some sand lenses are found in the clay, but none have been detected by drilling within the boundaries of the site. The mineralogical composition of the unit is quartz 12%, plagioclase 2%, carbonates 20%, and clay minerals 66%. Average composition

<403> CONT.
of the clays is illite 60%, montmoril. site 13%, and kaolinite and chlorite 27%. Permeability is on the order of 10(E-8) cm/s. Data on grain size, densities, water content, plasticity, etc., are provided. Laboratory determinations of the ion exchange capacity of the clays ranged from 10 to 50 meq/100 g. A heating experiment was conducted, but the results are questionable due to ineffective plugging of observation boreholes. Further heating studies are planned. Using estimated thermal properties, calculations show that in comparison with salt, clays require burial of smaller waste containers or older waste; also, for periods up to two years, no temperature increase occurs at a distance from the source greater than 15-20 m. Additional studies involving geological and engineering aspects of this site are planned. (LKN)

Calcium Carbonate Content; Clay Content; Grain Size Distribution; Moisture Content; Density; Ion Exchange Capacity; Plasticity; Permeability; Porosity

CLAYS; CALCIUM CARBONATE; FIELD STUDIES; LABORATORY STUDIES; GEOLOGIC STRATA; GEOLOGY; HEAT TRANSFER; ILLITE; MARLS; MONTMORILLONITE; REPOSITORY; SEDIMENTARY ROCKS; SEDIMENTS; SITE SELECTION; STORAGE, GEOLOGIC; STRATIGRAPHY; GEOLOGIC STRUCTURES; WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; PLASTICITY

<404>
Gera, F., and J.P. Olivier, OECD Nuclear Energy Agency, Paris, France.

Organization for Economic Cooperation and Development Countries Pursue Geological Disposal.

Nuclear Engineering International 23(266):35-37.(1978, January)

Nuclear waste disposal R and D programs are being embarked upon and expanded by 18 OECD countries. The authors emphasize the urgent need for an international demonstration program to allay political and public anxieties. Most OECD member countries with nuclear power programs are studying geological disposal. Disposal in either bedded salt or salt domes has been under consideration for more than 20 years. The U.S. and West Germany have performed most of the research and development work. Belgium, Italy, and the U.S. have ongoing programs aimed at the assessment of argillaceous formations. France, Japan, the U.K. and the U.S. have formed a Seabed Working Group to promote cooperation in assessing radioactive waste disposal into argillaceous sediments under the deep ocean floor. Hard rock research for geologic disposal is carried out by Austria, Canada, France, Sweden, the U.K. and the U.S. Finland, Spain and Switzerland are involved in a broad theoretical evaluation of hard rock formations existing within their respective territories. The emphasis of some of these programs is on finding suitable repository sites; in other cases the priority is on the design and execution of in-situ experiments. An experimental hard rock program is carried out in a granite formation at Stripa, Sweden in cooperation between Sweden and the U.S. Other in-situ experiments in hard rocks are being prepared in Canada and the U.K. (NAP)

WASTE DISPOSAL; STORAGE, GEOLOGIC; SEDIMENTS; GRANITES; REPOSITORY; SITE SELECTION; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; NIPP

<405>
Grover, J.R., K.D.B. Johnson, and J.B. Morris, United Kingdom Atomic Energy Authority, Atomic Energy Research Establishment, Harwell, England.

The Fixation of Radioactive Waste for Posterity.

Transactions of the American Nuclear Society 20:682-684; Nuclear Energy Maturity, Proceedings of the 1st European Nuclear Society Conference, Paris, France, April 21-25, 1975, (pp. 682-684).(1975)

This paper presents a general discussion of research on the controlled storage of high-level radioactive waste in the UK. The final pilot plant run of the FINGAL process for fixation of fission products in glass was in 1965, handling 19,500 Ci of liquid waste. The process combines evaporation, calcination of precipitated nitrates to oxides, and sintering of oxides to form glass in a stainless steel cylinder which also acts as the final storage vessel. The decision has now been made to shift from liquid to solid high-level waste storage in the UK, and emphasis is on the HARVEST project, (based on the FINGAL route) being carried out by BNFL. Estimated waste volumes expected to be produced per 1000 MW(e) yr. and treated by this process are 10 cu m of liquid waste from the Magnox reactor (low burnup fuel), yielding 3 cu m (0.5 MT) burnup fuel), yielding 3 cu m of glass (25% waste in the glass); from more advanced reactors (higher burnup fuel), the values will be 5 cu m of liquid waste or 1.4 cu m (% MT) of glass. Li/Ba borosilicate glasses presently appear to be the best, particularly for the high-Mg Magnox waste. Improved HARVEST processing might reduce fission product waste fixation costs to 0.25% of the value of the corresponding electricity. Problems still remaining with nuclear waste storage include that of H 3 production, the potential for separation and nuclear incineration of transuranics in waste after the other radionuclides have decayed, and the special problem of Pu arising from fuel fabrication processes. (LKN)

WASTES, HIGH-LEVEL; GLASS; VITRIFICATION; CALCINATION; WASTES, SOLID; WASTES, LIQUID; FIXATION; ECONOMICS; WASTE STORAGE; WASTE TREATMENT; PREDICTIONS

<406>
Grutzeck, M.W., Rockwell Hanford Operations, Richland, WA.

The Feasibility of Storing Radioactive Wastes in Columbia River Basalts.

Y/DW1/SUB-77/18268; CONF-770789; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 81-82), 92 pp.(1977, August 20)

The Atlantic Richfield Hanford Company has started a study to assess the feasibility of final geologic storage of radioactive waste

<406> CONT.

in deep caverns constructed in the Columbia River flood basalts. The project includes geologic studies, hydrologic testing, heat flow analyses, compatibility studies, and tectonic studies. These studies have led to the construction of a stratigraphic framework for the Pasco Basin, dividing the Yakima Basalt into four formations: the Saddle Mountain Basalt consisting of six basalt members, the Wanapum Basalt consisting of three basalt members, the Vantage Sandstone consisting of a very thin layer of indurated and well-sorted sandstone, and the Grande Ronde Basalt consisting of at least two stratigraphic units on the basis of marked differences in chemistry. The Umtanum Flow, a unit of the Grande Ronde Basalt, appears especially promising from a waste isolation standpoint. Its geologic characteristics and setting are favorable and its thickness exceeds 60 m within the Hanford Reservation. The hydraulic conductivity of all beds below the Vantage Sandstone is very small, especially for the dense basalts. Values for the Umtanum Flow are 3 X 10(E-6)-9 X 10(E-6) m/day. The flows proper appear to act as barriers separating the various interbeds. Geochemical studies now underway at Penn State University are fundamental to the understanding of the feasibility of long-term containment of radioactive waste and will require substantial additional research and development with emphasis on selection and characterization of the solidified waste form for storage and the compatibility of such form with the basalt host rock. Chemical reactions in wet as well as dry environments will be considered, and the effect of radiation on each component will be evaluated. All of this information will then be put together in a framework that will allow a full assessment of the viability of basalt as a final storage medium. (Auth) (LKH)

Only the abstract is given.

Hydraulic Conductivity; Stratigraphic Unit Thickness

BASALTS; IGNEOUS ROCKS; REPOSITORY; STORAGE, GEOLOGIC; GEOCHEMISTRY; HYDROLOGY; WASTES, RADIOACTIVE; WASTES, SOLID; RADIATION EFFECTS; GEOLOGY; HEAT EXCHANGE; TECTONICS; LABORATORY STUDIES; FIELD STUDIES; WASTE-ROCK INTERACTIONS

<407>

Hanstra, J., Reactor Centrum Nederland, Petten N.H., Netherlands.

The Waterproof Geometry of Salt Domes, An Important Safety Aspect for their Exploitation for Ultimate Disposal of High-Level Waste.

CONF-760310; STI/PUB/433; IAEA-SM-207/42; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, (pp. 295-302), 820 pp.(1976)

This report is a first attempt to demonstrate that virtually no water attack can create a local solution cavity in the flank or top of a salt dome as long as the dome is still in contact with its matrix bed. A safety analysis was made for a hypothetical salt dome which originated from a matrix bed of salt 3000-3500 m below surface and whose top is now 250 m below surface. Of this salt structure at least 200 m of rock salt was assumed to be maintained as a containment shield around both a dumping cavity for low- and intermediate-level wastes, and a burial horizon mined in this salt structure 800 m below surface for disposal of high-level wastes. This safety analysis led to the conclusion that only a continuous groundwater attack into either the top or flank of the structure could lead to containment failure in the long-term. The potential failure mechanism will not be the groundwater penetrating into the repository area, as long as the salt dome still remains in contact with its salt matrix. The buoyant force of the sediments overlying and surrounding the salt dome will, under certain stress conditions, enable salt to migrate towards the dissolving surface at such a rate that the salt dome will maintain its pierecement shape, notwithstanding the continuing dissolving action. If this salt migration should prove to be a containment failure mechanism, it may only affect the waste repository to a limited extent and after a very long period of time. It is this self-healing quality of the rock salt that establishes the waterproof geometry of salt domes. Model studies might reveal whether it is still realistic to adapt the layout of the waste repositories inside salt domes to the salt migration pathways. (Auth) (LKH)

SALT DOMES; GROUND WATER; DISSOLUTION; HALITE; EVAPORITES; SALT DEPOSITS; CAVITIES; HAZARD ANALYSIS; SAFETY; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, RADIOACTIVE; REPOSITORY; STORAGE, GEOLOGIC; WASTE STORAGE; WASTE DISPOSAL; THEORETICAL STUDIES

<408>

Hardaway, J.E., Isotopes, Westwood, NJ.

Possibilities for Subsurface Waste Disposal in a Structural Syncline in Pennsylvania.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 93-127), 253 pp.(1968, August)

Results of a combined literature and field study of the small structural syncline in the Dunning Cove area on the western perimeter of the central Appalachian Mountains of Pennsylvania suggest that the area may be suitable for injection of liquid wastes into deep permeable subsurface reservoirs. The syncline forms a basin just north of Bedford, PA, and occupies about 114.5 sq km (45 sq miles). The total thickness of Paleozoic strata is more than 2134 m (7000 ft). The stratigraphic section of greatest concern to this study includes the Upper Ordovician Reedsville Shale through the Middle Devonian Mahantango Formation of the Hamilton Group. The reservoir aquifer considered suitable for liquid waste disposal is the clean, quartzose Lower Silurian Tuscarora Sandstone. The investigation showed that the syncline has sufficient structural closure to warrant consideration as a reservoir. The Tuscarora crops out along two-thirds of the syncline's perimeter and is approximately 1128 m (3700 ft) beneath the center of the basin. It apparently is not faulted, and may be treated as a confined aquifer because it is underlain by the Juniata Formation (sandstone, siltstone, and shale) and overlain by the

<488> CONT.
Clinton Group shales, which probably have much lower permeability. Nineteen porosity measurements on 5 specimens of Tuscarora Sandstone from different zones gave an average porosity of 15%, with a range of 9 to 19%. Six porosity measurements on a specimen of the underlying Juniata Sandstone yielded an average of 10%, with a range of 9 to 12%. The probability of the presence of connate brine in the Tuscarora is enhanced by the fact that it is present in a deep well near Bedford. The Tuscarora strata are described as a "salaquifer" which will not promote harmful chemical interactions. Also, the fact that Tuscarora Sandstone is used for silica brick implies that its thermal conductivity increases with an increase in temperature - a benefit for heat dissipation of wastes. The Silurian Mills Creek and Clinton Group shales were studied and are regarded as impermeable barriers to migration of wastes and as favorable for the injection of waste/cement slurries into hydraulically produced fractures. However, grouting in shale appears less promising for large-scale injection than disposal in permeable sandstone. Detailed site evaluation and geologic, hydrologic, chemical, and physical tests should be performed on the Tuscarora before a final site is selected. (Auth) (DCW)

Though there is no specific mention of radioactive waste in this paper, the principles and geologic conditions apply, plus several references are on radioactive waste. The paper provides 99 references, 8 tables, 7 geologic and hydrologic maps, 18 isopach maps, 1 stratigraphic column, and 2 structural cross sections.

Porosity

WASTE DISPOSAL; WASTES, LIQUID; STORAGE, GEOLOGIC; GEOLOGY; GEOLOGIC FORMATIONS; GEOLOGIC STRATA; GEOLOGIC STRUCTURES; BASINS, STRUCTURAL; SEDIMENTARY ROCKS; SHALES; SANDSTONES; SLURRIES; GROUTING; HYDRAULIC FRACTURING; WELLS, INJECTION; HYDROLOGY; AQUIFERS, CONFINED; FIELD STUDIES; PERMEABILITY; POROSITY; SITE EVALUATION; SITE SELECTION; WASTE STORAGE; FRACTURES; REVIEWS; INJECTION

<489>
Haug, H.O., Kernforschungszentrum, Institut fur Heisse Chemie, Karlsruhe, German Federal Republic.

Final Purification and Concentration of Americium/Curium Separated from High-Level Reprocessing Waste.

Journal of Radioanalytical Chemistry 21:187-198; Proceedings of the 7th Radiochemical Symposium, Marianske, Lazne, Czechoslovakia, April 1973, (12 pp.). (1974)

A cation-exchange cycle has been developed for the recovery and concentration of the Am/Cm product from a DTPA/lactic acid solution used in an extraction process for the isolation and separation of the actinides from lanthanide fission products. The optimum pH region for sorption of Am^{3+} from 0.05M DTPA/1M lactic acid solutions by strongly acidic cation-exchange resins is pH 0.9-1.0. Maximum sample capacities, heights of the exchange zone, and concentration factors for different resins, cross-linkages and temperatures have been determined. Decontamination factors are given for some fission products as well as for U, Np and Pu. (Auth)

Table 5 shows decontamination factors during column loading with Pu.

ACIDS, ORGANIC; AMERICIUM; CATION EXCHANGE CAPACITY; CURIUM; CONCENTRATION FACTORS; DECONTAMINATION FACTORS; FISSION PRODUCTS; RESINS; ACTINIDES; LANTHANIDES; pH; TEMPERATURE; LABORATORY STUDIES

<490>
Neikes, G.H., and H.L. Bevier, Los Alamos Scientific Laboratory, Los Alamos, NM.

Petrology of Tuff Units from the J-13 Drill Site, Jackass Flats, Nevada.

LA-7563-MS; 55 pp. (1979, February)

The J-13 drill hole, located in Jackass Flats, Nevada Test Site, has penetrated 125 m of alluvium and 932 m of tuff. From top to bottom the stratigraphy is as follows: the Paint Brush Tuff, consisting of the Tiva Canyon Member - approximately 80 m thick, Tonopah Spring Member - 200 m thick, and an unnamed bedded tuff unit, about 95 m thick, composed of a complex of well-bedded air-fall and thin ash-flow layers, mostly unwelded; the Crater Flat Tuff, including the Prow Pass Member - approximately 50 m of welded tuff, a thinly bedded air-fall and unwelded ash-flow tuff, 12 m thick, possibly representing the base of the Prow Pass Member, and the Bullfrog Member - 250 m thick, composed of gray, biotite-rich ash-flow tuff with moderate to dense welding and several distinct cooling units; there is also a basal unnamed tuff below the Crater Flat - unwelded and lithic-vitric in composition. The well bottomed after penetrating 205 m of this unit. Most of the tuff deposits consist of welded tuffs; glass phases in the tuffs have been replaced by authigenic minerals, mainly K-feldspar, silica, and zeolites. The zonation of authigenic minerals, with depth, indicates that alteration of glass phases and filling of vugs occurred during welding and compaction of tuff units soon after deposition and by interaction with ground water. In nearly all of the tuffs, relict shard or pumice pyroclast shapes are preserved, even though the glass has been completely replaced. These relicts are apparently due to early alteration of a thin rim of the pyroclast to clay or deposition of submicrometer-sized grains of hematite. As alteration proceeds, a zeolite may be formed first, then eventually breaks down to form analcite or authigenic K-feldspar. It is also possible that several phases, for example K-feldspar, phillipsite, and silica, were formed simultaneously by interaction of glass with ground water. The zoning is similar to that described in core from Pahute Mesa, Nevada Test Site, and in tuff deposits of west Texas. This is reasonable, since Jackass Flats, although an intermontane valley, is bordered on three sides by fault blocks and has an outlet on the southwest corner, with a saturated aquifer of gentle hydraulic gradient toward the southwest. (Auth) (LKW)

Stratigraphic Unit Thickness

STRATIGRAPHY; LITHOLOGY; TUFFS; MINERALOGY; IGNEOUS ROCKS; VOLCANOES; GROUND WATER; FAULTS; DIAGENESIS; ZEOLITES; MORDENITE; FELDSPARS;

<810>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<810> CONT.
SILICA; FIELD STUDIES

<811>
Hite, R.J., U.S. Geological Survey.

Subsurface Geology of a Potential Waste Emplacement Site, Salt Valley Anticline, Grand County, Utah.

USGS OFR-77-761; 26 pp. (1977)

The Salt Valley anticline, which is located about 32 km northeast of Moab, Utah, is perhaps one of the most favorable waste emplacement sites in the Paradox basin. The site, which includes about 7.8 sq km, is highly accessible and is adjacent to a railroad. The anticline is one of a series of northwest-trending salt anticlines lying along the northeast edge of the Paradox basin. These anticlines are cored by evaporites of the Paradox Member of the Hermosa Formation of Middle Pennsylvania age. The central core of the Salt Valley anticline forms a ridgelike mass of evaporites that has an estimated amplitude of 3,600 m. The evaporite core consists of about 87 percent halite rock, which includes some potash deposits; the remainder is black shale, silty dolomite, and anhydrite. The latter three lithologies are referred to as "marker beds." Using geophysical logs from drill holes on the anticline, it is possible to demonstrate that the marker beds are complexly folded and faulted. Available data concerning the geothermal gradient and heatflow at the site indicate that heat from emplaced wastes should be rapidly dissipated. Potentially exploitable resources of potash and petroleum are present at Salt Valley. Development of these resources may conflict with use of the site for waste emplacement. (Auth)

Maps are given of Paradox basin showing salt inclines and drill holes. Gamma ray-neutron logs, bromine distribution in halite, and temperature profiles are illustrated.

Temperature

SALT DEPOSITS; EVAPORITES; FUELS; ROCKS; HALITE; BRONINE; POTASH; SEALES; DOLOMITE; ANHYDRITE; BOREHOLES; FAULTS; ROCK MECHANICS; WASTES, RADIOACTIVE; MAPS; LOGGING, GAMMA; THERMAL PROPERTIES; HEAT TRANSFER; GROUND WATER; CAP ROCK; FIELD STUDIES

<812>
Huckabee, W.A., Netherland, Sewell and Associates, Inc., Dallas, TX.

Regional Geology of the Northeast Texas Salt Dome Basin.

Y/OWI/SUB-76/99939; Geologic Study of the Interior Salt Domes of Northeast Texas Salt Dome Basin to Investigate Their Suitability for Possible Storage of Radioactive Waste Material, (pp. G-1 - G-21) 65 pp. (1976, May)

This regional study is part of an investigation of the tectonic and hydrologic stability of certain salt domes of the Northeast Texas Salt Dome basin. Records of approximately 1600 exploratory wells and 400 development wells were acquired and processed to provide data for subsurface studies of the basin. Structure maps of key horizons and isopachous maps of numerous stratigraphic intervals were constructed to determine the present configuration of the basin and its history of development. An extensive literature search was made to obtain additional geological and geophysical information about the study area; the literature on salt domes in general was also researched. The Northeast Texas Salt Dome basin is an embayment within the broad Gulf Coast basin. Major structural features adjacent to the present Northeast Texas basin are the Mexia-Talco fault system to the west and north, the Sabine Uplift to the east, and the Elkhart-Mount Enterprise fault system to the south. This framework encompasses a basin approximately 100 mi in diameter, with the center located near the city of Tyler. Prior to Jurassic time, the area had a complex geologic history that ended with deposition of thick Triassic red beds, block faulting, and erosion. As the basin sank, it became the repository of a thick section of Jurassic, Cretaceous, and Tertiary sediments. It is theorized that salt beds of the Louann Formation (Jurassic) near the base of this sequence became unstable early in the cycle of deposition; movements of the salt and overlying sediments created ridges, domes, and troughs that continued to develop as younger deposits accumulated in the sinking basin. The maximum relief of salt in the basin is about 22,000 ft. Net vertical movement of salt since the beginning of post-salt sedimentation (199 million yrs) has averaged about 0.065 mm/yr. Calculations of rates of uplift and thinning indicate the net rate of vertical salt movement was greatest during Late Jurassic time and has generally declined since. Estimated net average rates of uplift are 0.153 mm/yr in the Late Jurassic, 0.040 mm/yr in the Early Cretaceous, 0.020 mm/yr in the Late Cretaceous, 0.031 mm year during the Midway-Wilcox (Paleocene-Eocene) portion of the Tertiary, and 0.006 mm/yr during post-Wilcox time. The finding of this study indicate no significant upward movement of the salt at the present time. There are indications of some interior salt movement, confined primarily to replacement of salt being leached out by upwelling groundwater around and over some of the domes. (Auth) (LEH)

TECTONICS; HYDROLOGY; GEOLOGY; SALT DOMES; BASINS, STRUCTURAL; WELLS; CORES; MAPS; REVIEWS; GEOLOGIC STRUCTURES; STRATIGRAPHY; GEOLOGIC HISTORY; FAULTS; SEDIMENTARY ROCKS; DISSOLUTION; LEACHING; GROUND WATER; HALITE; EVAPORITES; CAP ROCK

<813>
International Atomic Energy Agency, Vienna, Austria.

Site Selection for Repositories of Solid High-Level and Alpha-Bearing Wastes in Geological Formations.

IAEA Technical Report Series No. 177; 64 pp. (1977)

This publication is the first attempt of the International Atomic Energy Agency (IAEA) to review information concerning the disposal of radioactive wastes in geological formations. The information given is intended as general

<813> CONT.

background for use by those authorities responsible for planning, approving and executing national radioactive waste management programs, and by those who review applications and make recommendations regarding repositories. Summary information is provided on many of the types of geological formations that might be considered by various countries as well as guidance on the geological and hydrological factors that should be investigated to demonstrate the suitability of the formations. The information was developed to the extent of current technology for application to the evaluation of deep geologic formations (greater than 365 meters below ground level). Areas not addressed in this review are: deep-sea disposal of radioactive wastes; introduction of high-level liquid wastes into geologic formations; conditioning of wastes for storage or disposal; and methods of constructing cavities. Conclusions by the agencies advisory group are as follows: 1) the general characteristics of geological formations support the validity of this disposal concept; 2) various types of geological formations and structures show promise as suitable media in which to locate repositories; 3) adequate long-term containment of radionuclides can be achieved through either the absence of circulating ground water or the combination of extremely slow fluid movements and high sorption capacity, particularly for long-lived alpha emitters, in some geological materials; 4) the plastic (self-healing) properties of certain geological formations also contribute to the reliability of the long-term containment of the radionuclides, as do other factors such as high tectonic stability; 5) each potential geological site for disposal of radioactive waste must be evaluated on its own merits; 6) due to the variability in characteristics of various long-lived radioactive wastes the two main categories, namely high-level and other alpha-bearing waste, may have different disposal requirements, particularly in relation to heat dissipation. (BIP)

REVIEWS, SITE SELECTION; WASTES, SOLID; WASTES, HIGH-LEVEL; ALPHA PARTICLES; GEOLOGIC FORMATIONS; GEOLOGIC STRUCTURES; WASTE DISPOSAL; REPOSITORY; GROUND WATER; SORPTION; CONTAINMENT; RECOMMENDATIONS; THERMAL PROPERTIES; WASTE STORAGE; SOLIDIFICATION; EVAPORITES; SALT DEPOSITS; SALT DOMES; ANHYDRITE; GYPSUM; POTASH; SEDIMENTARY ROCKS; SHALES; CLAYS; SANDSTONES; METAMORPHIC ROCKS; IGNEOUS ROCKS; GRANITES; BASALTS; TUFFS; TECTONICS; SEISMOLOGY; TOPOGRAPHY; FAULTS; JOINTS; CLEAVAGE; PERMEABILITY; POROSITY; DISPERSION; GASES; ROCK MECHANICS; MINERALS; SURFACE WATERS; BOREHOLES; EXCAVATION; SHAFTS; ECONOMICS; LAND USE

<816>

Jamison, D.B., and W.G.N. Cook, Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA; University of California, Department of Materials Science and Mineral Engineering, Berkeley, CA.

An Analysis of the Measured Values for the State of Stress in the Earth's Crust.

LBL-7071; SAC-07; 12 pp.(1978, August)

The state of stress in the crust of the earth is of great fundamental and practical significance. No totally satisfactory method for measuring the complete state of stress has been devised yet. Despite this, many efforts have been made to measure this state of stress at different locations. From a compilation of many of the results, fifty of which yielded the complete state of stress and in which one of the principal stresses is vertical, have been selected for a statistical analysis in an endeavor to define the nature of the state of stress in the crust. These data have been analyzed as a whole, and divided into three groups depending upon whether the vertical stress is the maximum, minimum, or intermediate principal stress. Linear regression analyses of the values of half the maximum stress difference as a function of half the sum of the maximum principal stresses have been made. The correlation coefficients for these fits are 0.786 for the data as a whole and 0.848, 0.790 and 0.383 for each of the groups. Values of the coefficient of sliding friction between blocks of rock comprising the crust, interpreted from the slopes of these lines, ranged from 0.625 (for those measurements where the vertical stress is the maximum principal stress) through 0.827 (for those cases where the vertical stress is the minimum principal stress), to 0.220 (for those cases where vertical stress is the intermediate principal stress). The 98 percent confidence limits for these values lie within +19.8 percent -16.6 percent. (Auth)

STRESS; STATISTICS; ROCKS; ROCK MECHANICS; FRICTION; FAULTS; THEORETICAL STUDIES

<815>

Japanese Governmental Agencies.

Oceanographic Research on Waste Disposal Off the Coast of Tokai-Mura.

A/CONF.15/P-1355; Peaceful Uses of Atomic Energy, Proceedings of the 2nd United Nations International Conference, Geneva, Switzerland, September 1-13, 1958. United Nations, Geneva, Switzerland, Vol. 18, (pp. 404-409).(1958)

Results of an oceanographic survey off Tokai-Mura prior to construction of a laboratory are reported. From current studies involving the Kuroshio and weaker coastal currents a counterclockwise circulation exists in an area south of Tokai-Mura at the western edge of the Kuroshio. Hence, when some substances are discharged into the sea in this area they are likely to drift southwards along the coast and spread over the area of circulation and ultimately be transported by the Kuroshio further out to sea. Sediment samples showed the activity is strongest for the mud fraction, and rather strong for the fine sand fraction. Chemical separation was performed to ascertain the nature of the nuclide in the sediments. The activity is large in the iron and alkali groups but is scarce in the calcium group. It appears that the nuclides are scavenged with the iron hydroxided flock. Diffusion experiments with dye indicated the value in the x-direction is 2113 sq cm/s. Tidal current diffusion coefficients are 10(E+8) for the x-direction and 10(E+5) to 10(E+6) for the y-direction. (NDV)

Grain Size Distribution

CONTAMINATION; DIFFUSION; DISPOSAL SITE; FIELD

<815>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<815> CONT.
STUDIES; HYDRODYNAMICS; OCEANOGRAPHY; RADIONUCLIDE MIGRATION; SEDIMENTS; SEA DISPOSAL; SEABED; WASTE MANAGEMENT; WASTES, RADIOACTIVE

<816>
Jenkins, I.L., United Kingdom Atomic Energy Authority, Harwell, England.

Problems in the Disposal of Irradiated Fuel Cladding.

AERE-M 8370; 43 pp.(1977, April)

This paper is concerned with problems likely to arise in the UK in dealing with stainless steel and Zircaloy residues from nuclear fuel processing. Problems considered include distribution of alpha emitters in stainless steel and Zircaloy hulls, special treatment for hull sections in which complete dissolution of fuel is difficult to achieve, selection of methods for densification of hulls, assessment of the feasability of pyrochemical and other processes for reduction of alpha emitter levels in hulls, and a shortage of information on properties of irradiated Zircaloy (embrittlement, pyrophoricity, criticality, Pu recovery, etc.). A review of published literature on high-temperature methods of dealing with hulls concludes that simple melting of stainless steel or Zircaloy gives good volume reductions but is unattractive because of the high temperatures required; melting of mixtures of Zircaloy and stainless steel is worth investigating as temperatures are lower (about 1100 C), volume reduction is likely, and some decontamination from ZO2 and FeO2 should be possible via filtration. Another method worth consideration is dissolution in molten zinc under a molten salt blanket. Other methods using chemical processing, such as fluorination or chlorination are more difficult and offer no significant advantages. A review of aqueous methods of treating fuel cladding determined that (1) the simplest method is "pickling" in a corrosive reagent, possibly aided by mechanical or ultrasonic agitation, and combined with a simple melting operation; (2) any complexing agent added to the reagent will likely cause difficulty in subsequent treatment; (3) dissolution of hulls in sulfuric acid, hydrofluoric acid, and various fluoride media is possible, but it will probably be necessary to add nitric acid to take up residues and to oxidize actinides to extractable valency states; (4) no extractant is likely to co-extract As(III) and the tetravalent actinides without also extracting Zr, Fe, Cr, and the rare earths; (5) extraction of actinides from HF and HNO3 mixtures needs to be examined; (6) electrolytic dissolution of stainless steel hulls to give a "pure" HNO3 liquor can probably be carried out; and (7) serious corrosion problems may arise in the storage of aqueous wastes produced in the extraction of actinides. (Auth) (LKH)

ZIRCALOYS; CLADDING; FUEL REPROCESSING; STEELS; WASTES, HIGH-LEVEL; WASTES, SOLID; WASTE TREATMENT; VOLUME REDUCTION; FILTRATION; DECONTAMINATION; ZINC; SALTS; FLUORINE; CHLORINE; COMPLEXES; ACIDS; EXTRACTION; ACTINIDES; ELECTROCHEMISTRY; CORROSION; WASTES, LIQUID; WASTE STORAGE

<817>
Johnson, K.S., University of Oklahoma, School of Geology and Geophysics, Norman, OK.

Evaluation of Permian Salt Deposits in the Texas Panhandle and Western Oklahoma for Underground Storage of Radioactive Wastes.

ORNL/SUB-4494/1; 145 pp.(1976, April)

The purpose of this project was to study the Permian salt deposits in and about 50,000 sq mi of the Texas Panhandle and western Oklahoma to determine whether the region contained any areas that might be suitable for geologic storage of radioactive wastes. The main basins investigated were the Palo Duro, the Delhart, and the Anadarko Basin. The principal parameters used in evaluating these three basins for radioactive waste storage include salt thickness, depth, tectonic and seismic history, lithology, permeability, proximity to aquifers, mineral resource content, extent of drilling and mining, and remoteness. A major part of this project involved a study of electric logs and some sample logs for about 300 wells drilled in the region. The results are of a reconnaissance type, but they do show the general character, thickness, distribution, depth, and structure of the salt deposits. The report concludes that the thick salt deposits of the Palo Duro Basin, and, to a lesser extent, those of the Delhart Basin, have many features that would be favorable for underground storage of radioactive waste. A number of factors detract from the suitability of similar deposits in the Anadarko Basin. (Auth) (LKH)

FIELD STUDIES; GEOLOGY; STRATIGRAPHY; SALT DEPOSITS; EVAPORITES; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; LOGGING, ELECTRIC; GEOLOGIC STRUCTURES; GEOLOGIC HISTORY; TECTONICS; AQUIFERS; BASINS, STRUCTURAL; LITHOLOGY; PERMEABILITY; MINERALS; MINING; SITE SELECTION

<818>
Jones, P.H., U.S. Geological Survey, Idaho Falls, ID.

Hydrology of Waste Disposal National Reactor Testing Station, Idaho, An Interim Report.

IDO-22082-USGS; 151 pp.(1961, December)

New tools and methods of investigation were required for the appraisal of the capacity of basalt aquifers and associated sediments to receive and attenuate radioactive waste. Borehole geophysics and modified oil-well testing techniques have been effectively employed at the National Reactor Testing Station (NRTS) in southeastern Idaho. Five prinipal aquifers were identifed and mapped in the Idaho Chemical Processing Plant (ICPP) area; their ability to transmit water was qualitatively evaluated, and correlated for distances of somewhat more than 4 mile, and their hydraulic head and quality of their water was determined. The effective thickness of the aquifers ranges from less than 1 foot to 18.5 feet. Structural maps of the altitude of their bases show relief ranging from 11 to 86 feet; principal directions of slope from the disposal well are southwest and southeast. Geophysical logs and chemical analyses of water collected by thief sampler indicated that cross-flow

<818> CONT.
between aquifers through the well bores was large, and was a very important factor in the waste-water movement in the ICPP area. Packers were installed to isolate and test selected aquifers in several wells, and head differences greater than two feet were observed between aquifers in a given well. To eliminate the large barometric fluctuations of water levels in the observation wells, the wells were fitted with air-tight caps and electronic water-level measuring equipment connected to high-speed recorders. In addition, each well was fitted with a submersible pump through air-sealed connections, so that samples of water could be obtained from an aquifer at selected times during a tracer test, for measurement of the flow velocity and mapping the travel path of labelled water under carefully controlled head conditions in the aquifer. The distribution of heat, caused by the discharge of warm water through the disposal well, and the distribution of salinity from the discharge of brines from water-softener regeneration, floor drain effluent, decontamination flushing water, and other waste waters to the disposal well were mapped by geophysical logging equipment. Results of these studies prove that the pattern of flow is more closely related to the geometry of the aquifer system than to the direction of the regional hydraulic gradient. Hydraulic data necessary for analysis of the factors of waste attenuation in this environment can be obtained, and detailed investigations are now in progress. (CAJ)

Report contains numerous figures, maps, and graphs.

Depth to Water Table; Stratigraphic Unit Thickness

ROCK MECHANICS; HYDRAULIC CONDUCTIVITY; HYDRAULIC HEAD; HEAD PRESSURE; HYDRAULICS; AQUIFERS; GEOPHYSICAL SURVEYS; WASTES, RADIOACTIVE; WASTE STORAGE; BASALTS; HYDROLOGY; WELLS; FIELD STUDIES; LOGGING, WELL

<819>
Keenan, T.K., Los Alamos Scientific Laboratory, Los Alamos, NM.

Waste Management Capabilities for Alpha Bearing Wastes at the Los Alamos Scientific Laboratory.

LA-UR-77-2686; CONF-771208; Treatment, Conditioning and Storage of Solid Alpha-Bearing Waste and Cladding Hulls, Proceedings of a Seminar, Paris, France, December 5, 1977, (11 pp.). (1977)

Waste Management activities at the Los Alamos Scientific Laboratory (LASL) involve a broad range of effort. There are requirements for daily processing of both liquid and solid radioactive and chemical wastes using a variety of technical operations. Approximately 0.5 x 10(E+7) l/yr of liquids and 9 x 10(E+3) cu m/yr of solids are processed by the Waste Management Group of the LASL. In addition, a vigorous program of research, development, and demonstration studies leading to improved methods of waste treatment is also carried out within the same group. The current developmental studies involve incineration of transuranic-contaminated combustible wastes as well as other waste management aspects of alpha emitting transuranic isotopes. Other studies in progress include a detailed examination of radiolytic reactions resulting from the interaction of the alpha particles of transuranic elements and hydrogenous waste matrices representative of those in retrievable storage at several U.S. sites in addition to the LASL. In addition to laboratory tests, actual waste-containing drums within the storage pits have been instrumented. (Auth) (NSF)

WASTE MANAGEMENT; ALPHA PARTICLES; WASTES, TRANSURANIC; WASTE PROCESSING; WASTES, LIQUID; WASTES, SOLID; WASTE TREATMENT; INCINERATION; HYDROGEN; LABORATORY STUDIES; FIELD STUDIES; DRUMS; PITS; INSTRUMENTS

<820>
Kolb, C.R., R.J. Lewis, and B.J. Miller, Louisiana State University, Institute for Environmental Studies, Baton Rouge, LA.

Quaternary Studies.

Y/OWI/SUB-4112/37; An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 353-426), 475 pp.(1977, September 30)

Studies of the Quaternary sediments were intended to aid in the determination of tectonic stability of the salt domes. Borings at Vacherie dome indicate no tectonic instability. Lows caused by dissolution and collapse of the underlying salt in pre-Quaternary times were found filled with oxidized fine sandy silts and silty sands about 200 ft thick. Lineations detected on radar, color, and black and white imagery show the accurate lineations on the eastern flank of the dome in the floodplain of Black Lake Bayou. This is the only suggestion of upward movement of the salt during the Quaternary. Vacherie and Rayburn's domes lie in topographic depressions above breached structural domes or anticlines. The most probable reason for these topographic depressions is that they are due to dissolution and collapse of the underlying salt. The magnitude of the erosional processes is predicted to be very small for the next 250,000 years. There is a possibility of the cap rock of Rayburn's dome becoming exposed during this time period. (RDV)

Diagrams illustrating the distribution of Quaternary sediments and soils are included. Appendices contain more data.

SALT DOMES; BASINS, DRAINAGE; DRAINAGE CHARACTERISTICS; FIELD STUDIES; GEOLOGY; HYDROLOGY; RIVERS; SOILS; SURFACE WATERS; TOPOGRAPHY; SEDIMENTS; EROSION; GEOLOGIC STRUCTURES; GEOGRAPHY

<821>
Krauhansl, J.L., Sandia Laboratories, Albuquerque, NM.

A Review of the Near Surface Heater Experiment at Oak Ridge, Tennessee

Y/OWI/SUB-77/18268; CONF-770789; Waste Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 85-86), 92

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<821> CONT.
pp.(1977, August 20)

An experiment has been undertaken at Oak Ridge to assess the large scale effects that heat from a waste canister would have were the canister emplaced in shale. The experiment will be conducted in the Conasauga Formation, a calcareous shale with interlayers of non-calcareous shales and siltstones. The experimental design includes a 13 ft long heater which will be buried at a depth of 55 ft and will run 6 months to a year at a surface temperature of 600 C, which should expel interlayer water in the clays and oxidize pyrite. The heater is surrounded by an array of thermocouples and stress gauges. In addition, coupons of potential canister materials are affixed to the base of the heater and will be examined after retrieval from the bottom of the heating element. Before and after the experiment the permeability of the formation will be measured using a Kr 85 tracer. Laboratory tests supporting the field test will be conducted. (LKB)

Only the abstract is given.

HEAT SOURCES; FIELD STUDIES; SHALES; SEDIMENTARY ROCKS; CONTAINERS; HEAT TRANSFER; STRESS; DESIGN; PERMEABILITY; WASTE DISPOSAL; WASTES, RADIOACTIVE; RADIATION EFFECTS; WASTE-ROCK INTERACTIONS

<822>
Nurfurst, P.J., T. Hugo-Persson, and G. Rudolph, Lawrence Berkeley Laboratory, Berkeley, CA; Hagby-Brak AB, Mora, Sweden; VIAK, AB, Falun, Sweden.

Borehole Drilling and Related Activities at the Stripa Mine.

LBL-7080; SAC-05; 47 pp.(1978, August)

Drilling operations for the Swedish-U.S. radioactive waste storage program at Stripa, Sweden were conducted from August 1977 to April 1978. One hundred sixty boreholes totalling over 2500 m were drilled. Two boreholes were diamond drilled from the surface to depths of 385 m and 360 m. Seven water table holes were dry drilled from the surface to 50-100 m depths. A pattern of over 350 subsurface boreholes were diamond drilled in the full-scale, time-scaled, extensometer, and ventilation drifts. Cores were from 38 mm to 406 mm in diameter and 4.5 to 18.5 m deep. All core from holes larger than 56 mm diameter was oriented. Seventy-six 38 mm instrumentation and heater boreholes were drilled to depths of 6-18 m using the Diamec 250 drill. Four 86 mm boreholes were drilled to depths of 11-13 m for the geophysical survey. Two 56 mm boreholes were drilled with the Diamec 250 and XP 60/90 drills to 5.5 m in the full-scale drift for use as pilot holes for the 406 mm main heater boreholes; 6 other 56 mm boreholes were drilled for the geophysical survey. Fifty-five 76 mm instrumentation and hydrological boreholes were drilled in the full-scale, time-scaled, extensometer, and ventilation drifts. Eight 127 mm heater boreholes were drilled in the time-scaled drift to depth of 11 m. Subsurface drilling rates ranged from 1.6 m/8 hr shift for the 127 mm boreholes to 10.7 m/8 hr shift for the 86 mm boreholes. Surface rates averaged 5.1 m/8 hr shift. Rate depended on size of the borehole, type of core barrel used (single, double, or triple tube), location of drill, orientation of the borehole, lithology, etc. Total drilling costs in Swedish krona/m ranged from 651 for the 38 mm boreholes to 1779 for the 127 mm holes. Core logging was done according to recommendations set forth by the Geological Society of London, 1970, with some modifications. Surveying results show that the specifications for accuracy and precision of drilling have been achieved and surpassed for all boreholes. Several new methods have been developed during the operation, including a smooth wall blasting technique to excavate large drift openings, techniques for drilling and recovery of 406 mm core, and slot drilling for 1 m and larger cores. Other operations conducted during this period included construction of surface installations, geophysical surveys, geologic mapping, stereophotography and fracture mapping in the drifts, TV logging, measurement of water influx to boreholes, geochemical water sampling, and pressure and injection tests. (LKB)

FIELD STUDIES; DRILLING; CORES; BOREHOLES; STORAGE, GEOLOGIC; ECONOMICS; METHODS; EXCAVATION; WASTES, RADIOACTIVE; GEOPHYSICAL SURVEYS

<823>
Lappin, A., Sandia Laboratories, Albuquerque, NM.

Studies of the Eleana Argillite.

Y/OWI/SUB-77/18268; CONF-770749; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, (pp. 47-49) 92 pp.(1977, August 23)

The Eleana Argillite of Mississippian and Pennsylvanian age is estimated to be 10,000 feet or greater in total thickness, and outcrops extensively on the Nevada Test Site. Abundant "clay-rich" argillite, with 15-80 wt% quartz, is restricted to the uppermost or "J" portion of the Eleana, and is much more restricted in extent. This part of the Eleana, since it is distinctly high in Al2O3 and FeO relative to an "average" shale, and low in Ca and K, is perhaps best thought of as a residual clay. Trace element contents of the argillite do not appear to be anomalous, however. The mineralogy of the 70 samples of Eleana examined thus far correlates well with the chemical data, and indicates metamorphism to temperatures of between 200 and 400 C. Further work is needed to obtain a better estimate of the maximum temperatures of metamorphism. Significant minerals identified include pyrophyllite, siderite, probable chamosite, chlorite, kaolinite, and a poorly-crystallized illite-vermiculite. The illite-vermiculite and siderite have been found to be stable for short periods of time at temperatures greater than 400 C. A major effort will be required to determine the long-term stability of all phases as functions of oxygen fugacity, water fugacity, and total pressure. Waste-rock interaction studies of the Eleana have become focused on its response to the emplacement of a near-surface heater. Thermal expansion to 500 C appears to vary as a function of both quartz content and orientation in even the most massive argillite, although the directional dependence of the massive type is

<823> CONT.

only 10%. Samples cut perpendicular to the bedding in a layered sample behave quite similarly to other samples up to 450 C, above which they contract, while samples cut in other orientations undergo a non-linear, and increasing expansion. Unconfined compression tests of massive argillite indicate a critical strength of up to 8300 psi. Work is being planned to determine thermal expansion, strength, and argillite elastic properties to temperatures of 500 C and pressures of up to 4200 psi. Heat capacities of powdered samples have been determined to 500 C, and show considerable variability as a function of mineralogy, and clay and water contents. When analyzed in air, one sulphide-rich sample goes through an exothermic sulphide oxidation at 425 C. Between 350 and 420 C, however, argillite heat capacities are fairly uniform and almost constant. Thermal conductivities at ambient temperature have been measured for more than 50 samples. Taken without regard to the layering, they average 2.69 W/m/deg/C. Conductivities measured parallel to the layering average 3.08 W/m/deg/C. Integrated thermo-mechanical modeling will begin at the start of FY 1978. (Auth)

Only the abstract is given.

Thermal Conductivity

SHALES; SEDIMENTARY ROCKS; THERMAL PROPERTIES; HEAT SOURCES; PRESSURE EFFECTS; THERMAL EXPANSION; HEAT CAPACITY; CLAYS; THERMAL CONDUCTIVITY; MODELS; MODELS, MATHEMATICAL; SITE SELECTION; REPOSITORY; STORAGE, GEOLOGIC; LABORATORY STUDIES; THEORETICAL STUDIES; WASTE-ROCK INTERACTIONS

<824>

Liljenzin, J.O., I. Svantesson, and I. Hagstrom, Chalmers University of Technology, Department of Nuclear Chemistry, Goteborg, Sweden.

A Possible Solution to the Long Time Storage Problem for High-Level Waste.

CONF-761020; Waste Management '76, R.G. Post (Ed.), Proceedings of a Symposium, Tucson, AZ, October 3-6, 1976, (pp. 303-312), 370 pp. (1979, October)

Actinide partitioning is suggested as a means of drastically reducing the volume of high-level waste which must be stored for long time periods. Reduction of Pu and Np levels in Purex waste by a factor of 10, Am and Cm by a factor of 50, and removal of all Cs and Sr, would reduce the hazardous period to about 100 yrs for the bulk of the waste. The separated Cs and Sr constitute the main heat and radiation sources in the HLW and require about 500 yrs storage. If the technology is developed, the separated actinides may even be burned in a fast reactor. The tentative flow sheet for actinide separation from Purex waste is as follows: HLW solution aged, heated, treated with nitrous gases, and filtered; using 1 M HDEHP solution in OK (Hysolvin 75 A) most of the U, Np, and Pu are removed, along with Zr, Nb, Mo, and Tc; the organic phase is then scrubbed with 6 M HNO3; high density polyethylene equipment would then be used to strip Zr and Nb with dilute HF; the extract is neutralized with 1 M ammonia and Mo, Tc, U, Np, and Pu are stripped with 1 M (NH4)2CO3; the organic phase is reacidified with nitric acid and recycled; the separated Zr and Nb together with the HNO3-waste solution is combined with the HLW, leaving the process for solidification. A further process step is needed to separate Mo and Tc from U and Pu. The U-Pu product could then be returned to the normal reprocessing plant. The excess nitric acid is removed by counter current extraction with 50% TBP in OK, then stripped with water, giving a roughly 2 M HNO3 product. The recovered acid contains some lanthanides and would have an estimated activity content of 0.2 Ci/l. After concentration it can be reused in the process. From the resulting low acidity raffinate it is possible to extract Am and Cm together with lanthanides using again a 1 M HDEHP solution in OK. The distribution ratios of Am and Cm are 50 and 45, respectively. The necessary separation of Am and Cm from the lanthanides can be made in several ways and this step has not been developed in detail as yet. The known extractants for Cs and Sr need a further decrease in the acidity of the HLW solution to be effective. This causes the formation of a voluminous precipitate containing a considerable amount of Cs and Sr which cannot be completely washed out. It is therefore considered necessary to remove several elements before the neutralization can be made. Development work is going on to find a suitable procedure. (Auth) (LKH)

ACTINIDES; PARTITIONING; WASTES, HIGH-LEVEL; PUREX PROCESS; HEAT SOURCES; RADIATION SOURCES; SEPARATION PROCESSES; NEPTUNIUM; PLUTONIUM; AMERICIUM; CURIUM; CESIUM; STRONTIUM; ORGANIC COMPOUNDS; ACIDS; RADIONUCLIDES; METHODS; WASTE TREATMENT; LABORATORY STUDIES; THEORETICAL STUDIES

<825>

Lomenick, T.F., Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

Terminal Storage of Radioactive Wastes in Geologic Formations.

CONF-760744; Energy Sources for the Future, J.L. Duggan, et al. (Eds.), Proceedings of a Symposium, Oak Ridge, TN, July 5-23, 1976. Oak Ridge Associated Universities, Oak Ridge, TN, (pp. 403-422), 422 pp. (1977, April)

The principal aim of the National Waste Terminal Storage (NWTS) program is to develop pilot plants and, ultimately, repositories in several different rock formations in various parts of the country. Rocks such as salt, shale, limestone, granite, schists, and serpentinite may all qualify as host media for the disposition of radioactive wastes in the proper environments. In general, the only requirement for any rock formation or storage site is that it contain any emplaced wastes for so long as it takes for the radioactive materials to decay to innocuous levels. The requirement, though, is a formidable one as some of the wastes will remain active for periods of hundreds of thousands of years and the physical and chemical properties of rocks that govern circulating groundwater and hence containment, are difficult to determine and define. Nevertheless, there are many rock types and a host of areas throughout the country where conditions are promising for the development of waste repositories. Rock salt has long been regarded as the most

<825> CONT.
favored geologic medium for the containment of radioactive wastes in the United States. This stems largely from the plastic behavior of salt which promotes self-healing and otherwise makes it impermeable to circulating water. Most of the attention has been directed towards the larger deposits, namely the Paradox and Permian basins, and those of the northeastern U.S. and the Gulf Coast region. Several thick sequences of shale and clay in the U.S. are favorable for further study as candidates for terminal storage facilities. In general, these deposits possess extremely low permeability, relatively high plasticity, high ion-exchange capacity, and are persistent, both vertically and laterally. Argillaceous rocks that are being studied include: 1) Late Ordovician shale in Ohio and New York, 2) Devonian-Mississippian shales in Ohio and Michigan, 3) Paleocene clays in Mississippi, Louisiana, and Arkansas, and 4) the Pierre shale and its equivalents in the upper Great Plains. Although calcareous rocks like limestone and dolomite are brittle and thus susceptible to extensive fracturing as well as to dissolutioning by circulating groundwater, these rocks may also be regarded as potential candidates for facilities in certain unique environments. One such environment is in northeastern Ohio where a deep dry mine has been developed in the Columbus limestone of Devonian age. Study areas for detailed investigations of crystalline and volcanic rocks are also selected. Significant accumulations of crystalline rocks occur in the Canadian Shield as well as in the Appalachian and Rocky Mountain orogenic belts. (Auth) (RAP)

STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; GEOLOGIC FORMATIONS; SALT DEPOSITS; SALT DOMES; REPOSITORY; SHALES; LIMESTONES; GRANITES; SCHISTS; CLAYS; ROCKS; BASALTS; TECTONICS; SEDIMENTARY ROCKS; ELASTOPLASTICITY; GROUND WATER; ANHYDRITE; POLYHALITE; EVAPORITES; POTASH; GEOLOGIC HISTORY; SEDIMENTS; EROSION; DENUDATION; STREAMS; LEACHING; DISSOLUTION; GASES; HYDRAULICS; BOREHOLES; STRATIGRAPHY; MINERALS; CORES; SCHEMOLES; THERMAL PROPERTIES; ROCK MECHANICS; PHYSICAL PROPERTIES; EARTHQUAKES; PERMEABILITY; ION EXCHANGE CAPACITY; EXCAVATION; GLACIERS; STRATIGRAPHY; REVIEWS

<826>
Rosenick, T.F., Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

Deep Exploratory Test Holes for Waste Repository Investigations.

Y/OWI/TM-8; 7 pp. (1976, September 14)

Since 1970 a total of 9 deep boreholes have been drilled by the AEC and its successor agency, ERDA, within six different study areas of the Permian Basin of Kansas and New Mexico to provide geohydrologic data that would aid in establishing the utility of the areas for the storage of radioactive wastes. Boreholes UCCBC-AEC #1 and #2 were drilled in 1970 at the Lyons, KS study area. Hole #1 was cored its entire length (1300.8 ft), while only the salt section from 739-1099.6 ft was cored in #2 hole. Geophysical logs, hydraulic data, and cores were also obtained from 3 other, privately drilled holes. All boreholes showed the salt deposits in the area to be structurally and stratigraphically sound; however, because of concern over nearby mining operations and a rather high density of boreholes in the area, and some negative political pressure, studies in the area were suspended. In 1972, 3 more boreholes were drilled, in two areas in Lincoln County, northwest of Lyons, and in western Kansas. These holes went to depths of 1100, 1275, and 2850 ft, respectively. Geohydrologic data from these holes indicated that the salt in the area was thinner than expected, and there were abnormally high concentrations of clay in the salt, prompting a continuation of the search for "better sites". In 1973-74, data collected from one private test well and three UCCBC-AEC boreholes in southeast New Mexico initially suggested the area might prove suitable, but then the presence of a localized "flow structure" in the salt horizon of one well was revealed. The latest borehole, drilled in an area a few miles southwest of the previous one, appears to indicate favorable subsurface conditions for a waste repository, although several more deep drill tests will be necessary. (LKN)

A table identifying exploratory boreholes cited in this report, including date completed, total depth, core information, logging, hydraulic test horizons, and potential emplacement formations is included.

BOREHOLES; HYDROLOGY; GEOLOGY; STORAGE, GEOLOGIC; REPOSITORY; CORES; GEOPHYSICAL SURVEYS; SALT DEPOSITS; CLAYS; SITE SELECTION; WASTE DISPOSAL; WASTE STORAGE; FIELD STUDIES

<827>
Rossnick, T.F., Oak Ridge National Laboratory, Oak Ridge, TN.

Geologic and Hydrologic Factors for Locating Repositories for Radioactive Wastes.

Southeastern Geology 21(1):75-82; Y/OWI/TM-14; 9 pp. (1979, December; 1977, February 01)

Rocks such as salt, shale, limestone, and granite may all qualify as host media for the disposition of radioactive wastes in the proper environments. In general, the only requirement for any rock body or storage area is that it contain any emplaced wastes for so long as it takes for the radioactive materials to decay to innocuous levels. This requirement, though, is a formidable one as some of the wastes will remain active for periods of hundreds of thousands of years, and the physical and chemical properties of rocks that govern circulating groundwater and hence containment, may be difficult to determine and define. Because of the complexities and variables of geologic environments, development of strict criteria for the selection of repository sites is not feasible. Rather, only general guidance for locating repositories is offered through a discussion of the following relevant factors: stratigraphy and lithology, structure, hydrology, topography, tectonics and seismicity, and physical and chemical properties of host rocks and potential waste/rock interactions. Thick beds of impervious rocks, especially plastic-behaving ones like shale which can deform without fracturing, would be good surrounding host rocks. Depending on the expense of evaluating and developing disposal space at great depths, possible plastic flow,

<027> CONT.
fracturing potential, and location of the water table, different depths of waste emplacement would apply for different host rock types. Structurally, best conditions would be stable, low-dipping sedimentary strata without faults or joints, with overlying protective impermeable beds. Repositories should be away from groundwater, but the distance depends on potentially changing flow regimes of surface streams and plasticity and healing capacity of rocks. Aquicludes should separate aquifers from brittle rocks. Topographically, low relief is good for surface transportation, reduces differential loading factors on subsurface salt formations, and implies possibilities of favorable subsurface geology. Low seismicity and tectonic stability are favorable factors. Physical and chemical properties of potential host rocks determine groundwater circulation, dissipation of radiodecay heat, radiation effects, and stability of mined openings. All of the above factors must be given careful examination at each individual site to determine waste-retention potential. (Auth) (DCW)

The report was submitted to Southeastern Geology, and was subsequently published with several paragraphs deleted.

WASTES, RADIOACTIVE; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; REPOSITORY; SITE SELECTION; SEDIMENTARY ROCKS; SHALES; SALT DEPOSITS; GEOLOGY; ROCKS; STRATIGRAPHY; LITHOLOGY; HYDROLOGY; TOPOGRAPHY; TECTONICS; SEISMOLOGY; STRUCTURAL STABILITY; GEOLOGIC STRUCTURES; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; WASTE-ROCK INTERACTIONS; GRANITES; IGNEOUS ROCKS; LIMESTONES

<028>
Martinez, J.D., R.L. Thoms, C.G. Smith, Jr., C.S. Kolb, R.J. Newchurch and R.E. Wilcox, Louisiana State University, Institute for Environmental Studies, Baton Rouge, LA.

An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes.

I/OWI/SUB-4112/37; 475 pp.(1977, September 30)

In order to determine the suitability of Gulf Coast salt domes a study of the tectonic and hydrologic stability of the domes was undertaken. All facets of geology, hydrology, geophysics and geochemistry were involved in the study. Growth rates for salt domes in the North Louisiana Salt Dome Basin were computed to be: Cenozoic, 0.3 mm/yr; Upper Cretaceous, 0.02 mm/yr; and Lower Cretaceous, 0.08 mm/yr. Preliminary findings indicate that no tectonic movement of consequence has occurred in the Quaternary. Monitoring systems (tiltmeters, laser ranging, precise leveling and engineering soil studies) are accompanying numerical modelling in determining vertical salt dome movement. Wells drilled and logged near the salt domes indicate that a significant amount of hydrologic instability is not evident. The discovery of the Cane River Formation more than 300 ft below the surface suggests subsidence following salt dissolution subsequent to Cane River deposition, 85 million years ago. Other zones of dissolution and collapse were found. Studies of other salt domes show no consistent trends in anhydrite concentration near the salt/caprock interface and euhedral crystals of anhydrite concentrate along and orient with salt boundaries. The hydrologic stability of salt dimes on the Gulf Coast was also assessed. Each chapter in the report is abstracted separately. (NDV)

The report has numerous diagrams and tables presenting important data and concepts.

CHEMICAL COMPOSITION; DISPOSAL SITE; BASINS, DRAINAGE; EQUATIONS; EROSION; FIELD STUDIES; GEOCHEMISTRY; GEOGRAPHY; GEOLOGIC FORMATIONS; GEOLOGIC STRUCTURES; GEOLOGY; GEOPHYSICAL SURVEYS; HYDROLOGY; MODELS, MATHEMATICAL; MINERALS; REPOSITORY; SALT DEPOSITS; SALT DOMES; SEDIMENTS; SEISMOLOGY; SOILS; WELLS

< 29>
McCann, T.P., E.C. Privrasky, F.L. Stead, and J.E. Wilson, Shell Canadian Exploration Company, Houston, TX; Tidewater Oil Company, Pittsburgh, PA; Magnolia, AR; Consolidated Gas Supply Corporation, Clarksburg, WV.

Possibilities for Disposal of Industrial Wastes in Subsurface Rocks on North Flank of Appalachian Basin in New York.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 83-92), 253 pp.(1968, August)

The north flank of the Appalachian basin in the state of New York was studied from well data, logs, literature, and any other available data, with the object of determining the suitability of the region for subsurface disposal of industrial (including radioactive) wastes, particularly liquid wastes. Permeable sandstone, salt beds that can provide leak-proof caverns, and shale that can contain grouted wastes in artificially produced fractures are especially significant. Subsurface strata dip southward at rates between 9.5 and 30.5 m/km (50 and 160 ft/mile). Local deformations such as low-relief anticlines and small displacement faults are few. Potential reservoirs for injection of liquids into permeable sandstone formations are the Cambrian Potsdam and Theresa Sandstones. The Potsdam has a maximum known thickness of 125 m (410 ft) and an average thickness of about 30.5 m (100 ft); the Theresa thickness ranges from zero to 457 m (1500 ft). Drilling depths to the Potsdam, the lowest potential reservoir, range from 305 m (1000 ft) to 3841 m (12600 ft). Other possible sandstone reservoirs are present in the Silurian and Devonian systems, but have less potential because of vagaries in porosity and permeability, because of the presence of unplugged or unrecorded wells in the shallower strata, and because of the presence of fresh water in the Upper Devonian. Silurian salt beds at depths between 153 m (500 ft) and 1220 m (4000 ft) offer sites for construction of storage cavities. Shale sections that appear to be suitable for storage of grouted wastes in hydraulically produced fractures are present in Upper Devonian and Upper Ordovician strata. There are thinner sections of possible interest for the same use in Silurian and Middle Ordovician rocks. This paper discusses each Cambrian, Ordovician, Silurian, and Devonian stratigraphic unit in detail, and summarizes

<829>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<829> CONT.
its lithology, areal extent, reservoir characteristics, and gas or salt water content; it also summarizes the regional physiography, stratigraphy, and structure. (Auth) (DCH)

This paper is primarily concerned with commercial wastes in general, but specific mention is made of radioactive waste disposal in salt caverns. The paper provides 149 references, 4 geologic maps, 7 structure contour maps, 5 isopach maps, 4 stratigraphic sections, 7 correlation charts, 2 tables, and 2 cross sections from wells.

WASTES, COMMERCIAL; WASTES, LIQUID; WASTES, SOLID; WASTES, RADIOACTIVE; WASTE STORAGE; WASTE DISPOSAL; BASINS, STRUCTURAL; GEOLOGY; GEOLOGIC STRUCTURES; GEOLOGIC STRATA; GEOLOGIC FORMATIONS; ROCKS; SEDIMENTARY ROCKS; SANDSTONES; SHALES; SALT DEPOSITS; CAVITIES; HYDRAULIC FRACTURING; GROUTING; PERMEABILITY; POROSITY; GROUND WATER; HYDROLOGY; SITE SELECTION; NATURAL GAS; FRACTURES; INJECTION

<830>
Hellen, F.F., Geological Associates, Jackson, MI.

Basal Ottawa Limestone, Chattanooga Shale, Floyd Shale, Porters Creek Clay, and Yazoo Clay in Parts of Alabama, Mississippi and Tennessee as Potential Host Rocks for Underground Emplacement of Waste.

Y/OWI/SUB-76/87950; 77 pp. (1976, February 28)

Impermeable rock units, preferably at least 500 feet thick and lying 1000 to 3000 feet below land surface, were sought in the region consisting roughly of the western 3/5ths of Tennessee and the northern 3/5ths of Alabama and Mississippi. All rock sequences, Cambrian through Eocene, were examined in varying detail, except the Cretaceous Selma Chalk and the diapiric salt. These rocks were studied for their relative impermeable homogeneity, their continuity, their background of structural and seismic stability and their hydrologic associations. The Central Mississippi Ridge of north-central Mississippi is overlain by a long-stable mass of Porters Creek Clay 500-700 feet thick, in an area roughly 50-60 miles wide and about 150 miles long. The Yazoo Clay, where best developed in the west-central and southwest part of Mississippi, is in the 400-500 foot thickness range, but locally exceeds 500 feet. The entire area mapped is underlain by the Louann Salt which has produced many deep-seated salt domes and numerous piercement salt domes. Salt flow has complicated shallow structural geology throughout that area. The Chattanooga Shale rarely exceeds 60 feet in thickness in the region studied and is generally much thinner and is absent in many places. In the lower part of the Middle Ordovician (Ottawa Megagroup), the Murphreesboro and associated dense limestones appear to offer a potential disposal unit 250-400 feet thick, having the advantages of rock competency and freedom from association with prolific aquifers in the overburden or beneath. Other less conspicuous stratigraphic units are reviewed. (Auth)

Sections of the document are illegible.

SITE SELECTION; ROCKS; STORAGE, GEOLOGIC; OUTCROPS; LIMESTONES; HYDROLOGY; SHALES; SEISMOLOGY; CLAYS; SALT DOMES; DIAPIRS; FAULTS; GEOLOGIC STRUCTURES; STRATIGRAPHY; GASES; LOGGING, WELL; MAPS; POROSITY; AQUIFERS; DOLOMITE; SANDS; GRAVELS; SEDIMENTS; FIELD STUDIES

<831>
Mercer, J.S., and B.R. Orr, U.S. Geological Survey, Albuquerque, NM.

Review and Analysis of Hydrologic Conditions Near the Site of a Potential Nuclear Waste Repository, Eddy and Lea Counties, Mexico.

USGS-OFR-77-123; 35 pp. (1977, February)

This interim report reviews and summarizes the hydrogeology associated with the Permian salt beds (Castile and Salado Formations) of Los Medanos area in southeast New Mexico. This information will be considered along with other factors in the preparation of an analysis of the impact of a potential waste repository on the environment. Sandstones of the Delaware Mountain Group, underlying the salt, have an average hydraulic conductivity of 0.16 ft/d, 15.665% average porosity, and groundwater flow velocities of 0.0005-0.0008 ft/d north-northeastward toward the margin of the Capitan Limestone. The Capitan is a reef aquifer adjacent to the study area. It has transmissivity values from 500-10,000 sq ft/d and an average hydraulic conductivity around 5 ft/d, with groundwater movement west of the Pecos River in a northeastward direction, discharging at Carlsbad Springs; east of the Pecos, water movement is slow and direction uncertain. The Castile and Salado Formations (prospective repository horizons) have a few isolated brine and gas pockets, but transmit little or no groundwater. Dissolution of the top of the Salado in some localities has produced an evaporite residuum in which brines move southward along Nash Draw and discharge into the Pecos at Malaga Bend (discharge = 200 gal/min.) Transmissivity of the residuum is 8000 sq ft/d and flow rate is 0.2 ft/d. Above the Salado is the Rustler Formation; water bearing units in it are the Culebra Dolomite Member and the Magenta Member (discharge up to 700 gal/min). Transmissivity in the Rustler is from 0-500 sq ft/d; with movement south-southwestward toward the Pecos. Ground water in the Dewey Lake Red Beds is restricted to sand lenses; movement is very restricted. The Santa Rosa Sandstone (Triassic) lies above the Dewey Lake [illegible] has low permeability; porosity is ab[illegible] with flow 0.3 ft/d south-[illegible]st toward the Pecos. Data on the Chi[illegible]ation, also Triassic, is sparse; lithology (mudstone) indicates it would be a poor aquifer. Other potential aquifers near the proposed repository area include the Ogallala and Gatuna Formations (Cenozoic), and alluvial deposits. Most water in the Ogallala moves to the southeast but some moves westward, recharging the Santa Rosa Sandstone aquifer. The Gatuna contains water in sand and gravel lenses. Groundwater movement in the alluvium is generally less than 1 ft/d; transmissivity is 13,600 sq ft/d. (Auth) (LKM)

Hydraulic Conductivity; Porosity; Hydraulic Velocity; Transmissivity

SALT DEPOSITS; STRATIGRAPHY; GEOLOGY; HYDROLOGY; WIPP; SANDSTONES; LIMESTONES; AQUIFERS; REPOSITORY; GROUND WATER; EVAPORITES; BRINES;

<031> CONT.
DOLOMITE; ALLUVIUM; REVIEWS; SITE SELECTION

<032>
Huehlberger, H.R., Hetherland, Sewell and Associates, Inc., Dallas, TX.

Genesis and Growth of Salt Domes in the Northeast Texas Salt Dome Basin.

E/ONI/SUB-76/99939; Geologic Study of the Interior Salt Domes of Northeast Texas Salt Dome Basin to Investigate Their Suitability for Possible Storage of Radioactive Waste Material, (Rp. 2-8) 65 pp.(1976, May)

This report is a general explanation of salt dome genesis and growth using the Northeast Texas Salt Dome basin as an example. Probable causes of salt dome formation are: 1) differential deposition of sediments resulting in uneven loading which causes salt to migrate from areas of maximum stress; 2) basin tilt of salt and sediments along basin edges, causing up-dip "pull-apart" structures and down-dip creep and folding; 3) buoyancy due to density differences between the relatively "light" salt and overlying "heavy" compacted sediments; and 4) other factors such as horizontal compression, compaction, etc. The salt tends to migrate toward areas of minimum stress, where it forms a diapir (salt plug or stock) about 1-2 mi in diameter, which moves upward as it is fed by lateral salt flow. It is generally accepted that salt dome movement in most cases is via a series of sporadic, non-uniform uplifts. This uplift is increased by rapid deposition of sediments in a sinking basin such as the present Gulf Coastal area and is diminished, or halted altogether, during periods of nondeposition or regional emergence such as exists in the Northeast Texas area. Upward growth of a salt dome may be halted by: 1) depletion of replacement salt at the base of the dome; 2) resistance encountered in the overlying sediments, including drag between the diapir and surrounding sediments; 3) buoyancy neutralization as the salt penetrates shallow, uncompacted low-density beds; or 4) dissolution of the top of the dome by a freshwater regime, which causes accumulation of a massive anhydrite caprock that offers further resistance to upward growth. Rates of salt dome growth can be derived from deformation of time/stratigraphic markers by the moving diapir, from the structure and stratigraphy of beds overlying the dome, and by the size of the rim syncline at various stratigraphic levels. In the case of caprock formation in the presence of fresh water, if the anhydrite/salt ratio in the dome can be assumed to be the same as in seawater, then each foot of caprock represents 30 ft of dissolved salt dome, and indicates the amount of vertical movement. The history of sediment deposition in the Jurassic and Early Cretaceous indicates that deltaic sedimentation, with accompanying uneven loading, existed in the Northeast Texas basin. Basin tilting resulted on down-dip creep of sediments and salt; as the basin sank and sediment thickness increased, an unstable buoyancy condition between the salt and overlying sediments was established. Salt dome growth began when this instability reached a critical level. Geologic studies indicate gradually decreasing rates of salt dome uplift in the Jurassic through Early Cretaceous, the Late Cretaceous, the Midway and Wilcox Groups (Early Tertiary), and post-Wilcox beds, as the centers of rapid sedimentation shifted southward toward the present Gulf Coast. At present, uplift rates for the domes of the Northeast Texas basin appear to have slowed to a minimum or halted altogether. (Auth) (LKH)

SALT DOMES; EVAPORITES; ANHYDRITE; GEOLOGY; GEOLOGIC STRUCTURES; GEOLOGIC HISTORY; BASINS, STRUCTURAL; REVIEWS; HALITE; DISSOLUTION; GROUND WATER; HYDROLOGY; CAP ROCK; SEDIMENTARY ROCKS

<033>
National Aeronautics and Space Administration, Lewis Research Center, Cleveland, OH.

Feasibility of Space Disposal of Radioactive Nuclear Waste. 2. Technical Summary.

NASA TM X-2912; 119 pp.(1974, May)

The feasibility of transporting radioactive waste from nuclear power plants into space for ultimate disposal was investigated at the request of the Atomic Energy Commission (AEC) as a NASA in-house effort. The investigation is part of a broad AEC study of various concepts for long-term storage or disposal of radioactive waste. Both expendable and reusable space vehicles were considered for the space disposal missions. The Space Shuttle in conjunction with either reusable or expendable space tugs provided the lowest cost per payload for delivery to the various destinations investigated. The choice of space destinations was narrowed to high earth orbits, solar orbits, and solar system escape. The latter destination appears to be the most desirable and was found to be economically and technically feasible. The nuclear waste packages were designed to prevent loss of radioactive waste in accident environments. Several compositions of radioactive nuclear waste were considered: fission products, and actinides containing various percentages of fission products. The long-lived radioactive isotope group referred to as actinides (heavy isotopes, with atomic numbers above 89), which could be a separate part of the waste, received the greatest attention. The disposal of only this relatively small quantity of the most dangerous components of nuclear waste appears to be cost effective. The results of this exploratory study indicate that disposal into space of the long-lived actinides of nuclear waste appears feasible, from both the economic and safety viewpoints; space disposal of all fission products does not appear practical. The transportation costs for ejecting the actinides out of the solar system would represent less than a 5-percent increase in the consumer bill for electric power generated by nuclear power plants. Such missions involve certain risks, however small, which would have to be balanced against the benefits to be derived from removing the dangerous long-lived radioactive waste from man's environment and relieving future generations from the responsibility of protecting themselves against our radioactive waste. Firm plans for such nuclear disposal missions must be based on more study, development, and testing. (Auth) (NAP)

See also NASA TM X-2911, Executive Summary.

SPACE TECHNOLOGY; WASTES, RADIOACTIVE; SPACE DISPOSAL; COST BENEFIT ANALYSIS; SOLAR SYSTEM; ACTINIDES; FISSION PRODUCTS; SAFETY; ACCIDENTS;

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GEOLOGY, HYDROLOGY AND SITE RESOURCES

<033> CONT.
WIRES; CONTAINERS; PACKAGING; SHIELDING; THERMAL PROPERTIES; STORAGE, EXTRATERRESTRIAL

<036>
National Aeronautics and Space Administration, Lewis Research Center, Cleveland, OH.

Feasibility of Space Disposal of Radioactive Nuclear Waste. 1. Executive Summary.

NASA TM X-2919; 18 pp. (1973, December)

The concept of disposing of radioactive waste into space was studied and found to be feasible. Tentative solutions are presented for technical problems of safely packaging the separated long-lived actinide wastes. Disposal of these wastes is the primary concern because they will remain radioactive for extremely long times. The package design includes shielding to achieve reasonably low external levels of radiation. The logistics and potential hazards of launching these packages into either high Earth orbits or solar orbits or to escape the solar system have been evaluated. These destinations have been found to be the most promising. Although the solar system escape requires greater energy, it appears to be the most desirable for ultimate disposal. The total costs of a system for space disposal of radioactive waste are based on the rate of accumulation of fission products and uranium-free actinides in reprocessing plants serving the nuclear power industry and on the launch costs, the destinations, and the launch frequency. The number of waste packages to be launched per year depends on the degree of separation of the long-lived actinides. For example, a package containing about 200 kilograms of separated actinide wastes with about 0.1 percent residual fission products could be ejected out of the solar system for a cost of about $150,000 per kilogram. Fifty to 100 Space Shuttle launches of such packages per year would be required in the 1990-1995 time period to handle the actinide waste. To this cost must be added the estimated cost of separating and encapsulating the actinide waste. Although the space transportation cost would be several billion dollars per year, the cost prorated over the nuclear electrical capacity is less than 0.1 cent per kilowatt-hour. A packaging design concept has been evolved that appears on a qualitative basis to provide protection against the radioactive waste in accident environments. The concept, however, does need a follow-up experimental program and safety assessment to establish a system design. (Auth)

See also NASA TM X-2912, Technical Summary.

SPACE DISPOSAL; WASTES, RADIOACTIVE; PACKAGING; DESIGN; ACTINIDES; SITE SELECTION; HAZARD ANALYSIS; SAFETY; SOLAR SYSTEM; COST BENEFIT ANALYSIS; FISSION PRODUCTS; STORAGE, EXTRATERRESTRIAL; SPACE TECHNOLOGY

<037>
National Research Council, Commission on Natural Resources, Committee on Radioactive Waste Management, Panel on Geological Site Criteria, Washington, DC.

Geological Criteria for Repositories for High-Level Radioactive Wastes.

19 pp. (1978, August 3)

The study considers the use of a carefully engineered structure located deep underground in a suitable geological environment as the most feasible method of long term, terminal isolation of high-level radioactive wastes. General criteria are developed for evaluating the suitability of geological environments such as: long term stability, geometrical and dimensional criteria, hydrological and geochemical criteria, and geo-economic factors. (CAJ)

GEOPHYSICAL SURVEYS; GEOLOGY; SITE SELECTION; HYDROLOGY; REPOSITORY; GEOCHEMISTRY; STORAGE, GEOLOGIC; WASTE STORAGE; WASTES, HIGH-LEVEL; ROCK MECHANICS

<038>
National Technical Information Service, Springfield, VA.

Radioactive Waste Disposal in Salt Deposits - A Bibliography with Abstracts.

NTIS/PS-78/1238; 168 pp. (1978)

This updated NTIS bibliography cites government sponsored research reports on underground disposal of radioactive wastes in salt deposits published between 1964 and November 1978. It contains 168 abstracts, 77 of which are new entries to the previous edition. The citations cover radiation protection, disposal efficiency, site selection, site stability, mathematical models, and hydrology. (Auth) (LKN)

Supersedes NTIS/PS-77/1085.

BIBLIOGRAPHIES; WASTE DISPOSAL; WASTE STORAGE; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; SALT DOMES; SITE SELECTION; HYDROLOGY; MODELS, MATHEMATICAL; METHODS; SAFETY; RADIATION HAZARDS; STRUCTURAL STABILITY

<037>
Nelson, P., B. Paulsson, R. Machiele, L. Andersson, T. Schrauf, W. Rustrelid, O. Duran, and K.A. Magnusson, Lawrence Berkeley Laboratory, Berkeley, CA; Terra Tek, Salt Lake City, UT; Swedish Geological Survey, Uppsala, Sweden.

Preliminary Report on Geophysical and Mechanical Borehole Measurements at Stripa.

LBL-8280; 165 pp. (1979, May)

Four different systems of borehole instrumentation have been employed thus far at an experimental site at Stripa, Sweden where a series of hydrological and thermomechanical experiments are being conducted to examine the suitability of granitic rock for the storage of radioactive waste materials. A suite of 7 logs -- neutron, gamma-gamma, resistivity, gamma ray, sonic, caliper, and temperature -- operated in a borehole (SBH-1) of 300 m depth located 11 zones where the rock permeability is expected to be enhanced due to the presence of open fractures. The sonic waveform record

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proved especially useful in this regard. Below 330 m slant depth (230 m true depth) indications of fracturing decrease considerably and the rock appears homogeneous and intact. The planned injection testing of SHM-1 will test the utility of the log interpretation. Borehole measurements were also acquired in a large number of boreholes 5 to 14 m in length located in experimental drifts 340 m underground. Here several physical properties, including porosity, density, sonic velocity, and borehole ragosity, are generally quite uniform, with the exception of a few local chloritic zones and a few minor fractures. However, in situ determinations of the mechanical modulus (elastic modulus) with the CSM cell indicate substantial variability, with some apparent fracture control. The modulus drops from 45 GPa to 33 GPa as fracturing increases; standard deviation of the measurement is typically 10 GPa. With the exception of a few anomalous zones, the neutron log indicates that total water content does not vary by more than 1 vol %. From core analysis, the total water content in the Stripa granite is 2.4 + or - 0.5 vol %. Uranium and thorium concentrations are quite high in the Stripa granite, with local fluctuations associated with chloritic zones as revealed by the gamma-ray log. Dramatic increases in the gamma-ray log in a few holes are associated with higher rates of water inflow carrying Rn decay products into the holes. A differential resistance probe appears promising as a sensitive detector of fine fracturing. A cross-hole ultrasonic (about 30 kHz) propagation system indicates variations of a few percent in compressional- and shear-wave velocities, reflecting the presence of fractures and changes in fracture characteristics as the rock is heated in a simulated storage test. The geophysical and mechanical data are being compared with the results from core and television logging, with hydrological test data on static pressure and injection permeability, and with displacements induced by thermal loading. (Auth) (LKH)

Graphic and tabular data are provided in the text and in appendices.

Moisture Content; Elastic Modulus

BOREHOLES; INSTRUMENTS; LOGGING, GAMMA; LOGGING, NEUTRON; LOGGING, SONIC; LOGGING, RESISTIVITY; LOGGING, CALIPER; LOGGING, TEMPERATURE; FRACTURES; PERMEABILITY; IGNEOUS ROCKS; GRANITES; POROSITY; DENSITY; MOISTURE CONTENT; PHYSICAL PROPERTIES; URANIUM; THORIUM; CHLORITES; GROUND WATER; RADON; DAUGHTER PRODUCTS; RESISTIVITY; THERMAL PROPERTIES; GEOPHYSICAL SURVEYS; SITE SELECTION; SITE EVALUATION; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC

<038>
Netherland, Sewell and Associates, Inc., Dallas, TX.

Geologic Study of the Interior Salt Domes of Northeast Texas Salt Dome Basin to Investigate Their Suitability for Possible Storage of Radioactive Waste Material.

Y/OWI/SUB-76/99939; 65 pp. (1976, May)

The purpose of this study of the salt domes of the Northeast Texas basin was 1) to investigate the movement and hydrologic stability of these domes, 2) to identify one of more of these domes which is suitable for further study as a potential radioactive waste repository, and 3) to outline the additional information needed to more thoroughly evaluate these domes. An extensive search of the pertinent literature, including records of approximately 1600 exploratory wells and 400 development wells, as well as geophysical data, indicates that growth of the salt domes in the basin has slowed with geologic time, and is presently minimal or nonexistent. Hydrologic data indicates that the Ballard, Whitehouse, and Keechi domes are not currently subject to significant salt dissolution by fresh water, and can for practical purposes be considered hydrologically stable. However, to more thoroughly evaluate this stability it is recommended that 1) a series of shallow water wells be drilled to determine if dissolution is occuring and in what quantities, 2) surface waters and springs be sampled for evidence of salt dissolution and transport to the surface, and 3) core samples of caprock and salt be analyzed to determine chemical and physical characteristics of the caprock and the percent of insoluble impurities in the salt (correlating caprock formation with rate of salt dissolution). It is suggested that evidence of geologically recent salt dome movement be investigated by study of near-surface alluvium and terrace deposits over or adjacent to the domes, and that the size, shape, and configuration of the domes be more precisely defined via core hole, gravity, and seismic data. Finally, it might also be of use to obtain information on 1) the presence of caprock draped over the flanks of some of these domes, 2) the more precise definition of the rates and patterns of groundwater flow near the domes and throughout the basin, 3) the physical reactions taking place at the salt-caprock interface, and 4) other phenomena associated with water dissolution, caprock formation, dome growth, vertical salt movement, etc. (Auth) (LKH)

SALT DOMES; BASINS, STRUCTURAL; GEOLOGY; HYDROLOGY; SITE SELECTION; REPOSITORY; EVAPORITES; CAP ROCK; GEOLOGIC STRUCTURES; GEOLOGIC HISTORY; REVIEWS; HALITE; DISSOLUTION; GROUND WATER; SEDIMENTARY ROCKS; SURFACE WATERS; STRUCTURAL STABILITY; FIELD STUDIES; TECTONICS; STRATIGRAPHY; STORAGE, GEOLOGIC; WASTE STORAGE

<039>
Netherland, Sewell and Associates, Inc., Dallas, TX.

Geologic Investigation of the Virgin River Valley Salt Deposits, Clark County, Southeastern Nevada to Investigate their Suitability for Possible Storage of Radioactive Waste Material as of September, 1977..

Netherland, Sewell and Associates, Inc. Report; 62 pp. (1977, September)

The Virgin River Valley salt deposits, Clark County, southeastern Nevada, have been investigated to evaluate their suitability for further study in relation to possible storage of radioactive waste material. Salt deposits of the Virgin River Valley basin are part of the Pliocene(?) Muddy Creek Formation, a relatively flat-lying sequence of non-marine sediments bounded on the east and west by complexly folded and faulted

<039> CONT.
pre-Cenozoic rocks. Salt is exposed at the surface in several places; elsewhere it underlies up to 2256 ft of overburden. Known salt thickness ranges from 0 to 1750 ft. Most of the salt appears to be relatively pure (93% NaCl). There is considerable evidence of local distortion of the salt in the part by folding, faulting, shearing, and possibly diapirism. Over half of the salt body underlies the Overton Arm of Lake Mead to the east; the dry land portion of the body having a thickness of at least 1000 ft is estimated to cover an area of about 8.5 sq mi. Current tectonic activity in the area is believed to be confined to crustal adjustments following the filling of Lake Mead, although fault movement along the Las Vegas Wash zone has been dated as recently as 10,000 yrs ago and there is evidence of activity along the fault that bounds the west side of the Valley almost to the present. Detailed information on the local hydrology is not presently available but it is reported that a groundwater study by the U.S. Geological Survey is in progress. There is no evidence of exploitable minerals in the area other than evaporites and possibly sand and gravel. Additional information is needed to throughly evaluate the salt deposits of the Virgin River Valley. This includes: (1) definition of the configuration, composition, and internal structure of the deposits via coring and geophysical surveys; (2) more thorough evaluation of salt dissolution and the hydrologic stability of the salt deposits; (3) a more detailed study of the tectonic stability of the salt deposits and surrounding area, including seismic monitoring; (4) determination of whether or not approval could be obtained from the federal, state, and local agencies in charge of the Lake Mead Recreation Area (on which the deposits are located) for use in the storage of radioactive wastes; and (5) evaluation of other salt deposits in the general region of southern Nevada and northern Arizona several of which are discussed briefly. (LKH)

SALT DEPOSITS; WASTE STORAGE; STORAGE, GEOLOGIC; GEOLOGIC STRUCTURES; FRACTURES; FAULTS; EVAPORITES; STRUCTURAL STABILITY; TECTONICS; SITE SELECTION; REPOSITORY; GEOLOGIC HISTORY; DISSOLUTION; FIELD STUDIES

<040>
Parimeau, W.G., RE/SPEC, Inc., Rapid City, SD.

Thermoelastic/Plastic Analysis of Waste-Container Sleeve: 3. Influence of Salt Strength on Sleeve Loading.

CNWL/SUB-0269/2; 18 pp.(1975, March 21)

This report represents the results of waste container/sleeve loading estimates as obtained by axially symmetric thermoelastic/plastic analysis using salt tensile strengths that are considerably less than the compressive strengths. Three combinations of salt tensile, compressive and shear strength in linear and non-linear yield conditions used in the estimates show no influence on the analysis of the New Mexico repository concept. The salt remains elastic throughout the excavation and subsequent 10 year heating period. Tensile stresses are not observed. Tensile strength is thus not important to the analysis even at 10% of the compressive strength value. Although strictly applicable only to the conditions of the analysis reported here, the capability for incorporating arbitrary strength combinations in linear or non-linear yield conditions is demonstrated. Computer plots of principal stresses and displacement fields at various stages of the excavation and heating simulation aid in the visualization of repository concept mechanics and show the possible need for additional mesh refinement for more precise stress information. (Auth) (LKH)

VIPP; MODELS; MODELS, MATHEMATICAL; STRENGTH, TENSILE; STRENGTH, COMPRESSIVE; STRENGTH, MECHANICAL; ROCK MECHANICS; SALT DEPOSITS; REPOSITORY; STORAGE, GEOLOGIC; ELASTOPLASTICITY; THERMAL PROPERTIES; STRESS; EXCAVATION; DESIGN; THEORETICAL STUDIES

<041>
Paterson, W.S.B., Canadian Department of Energy, Mines and Resources, Ottawa, Ontario, Canada; Polar Continental Shelf Project.

Radioactive Waste Disposal in Antarctica: Some Glaciological Aspects.

Modern Geology 6:37-42.(1976)

Radioactive waste disposal in the Antarctic ice sheet is evaluated, based on a review of pertinent literature. Movement of the ice sheet, and therefore of any materials contained within it, is by two mechanisms: 1) deformation of the ice itself, or 2) sliding of the ice over its bed. If the ice in contact with bedrock is at its pressure melting point, the water generated will tend to flow towards the edge of the ice sheet at a relatively fast rate; thus radioactive materials should not be dumped in any region where there may be water at the bed. The problem of glacial surges must also be considered: Although the mechanism of surges is not understood, it is known that the larger the glacier, the longer the time interval between surges; for Antarctica, surges might be expected at intervals of thousands of years, if ever. There is some evidence of a disintegration of the West Antarctic ice sheet during the Pleistocene, possibly due to a surge. If the ice sheet were to surge, radioactive waste deposited far inland might be carried to the sea in a matter of years. Proposals to allow waste canisters to melt their way through the ice to bedrock are viewed as unwise, since they might create instability and initiate a surge. An alternative would be to arrange for the canisters to follow the slow downward path traced by ice particles deposited near the surface; in this way, they would not reach bottom for perhaps 20,000 years. Before a decision on dumping waste in the ice sheet can be made, more work must be done on the effect of water at the ice/bedrock interface on flow, the cause and mechanism of surges, the circumstances in which temperature instability may be important, temperature values within the ice sheet and at the bed, and computer modeling of the ice sheet in a non-steady state, including effects of any conceivable climate change. (LKH)

GLACIERS; WASTE DISPOSAL; WASTES, RADIOACTIVE; CANISTERS; THERMAL PROPERTIES; RECOMMENDATIONS; REVIEWS

442>
Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Waste-Rock Interactions.

Y/OWI/SUB-77/14260; CONF-770789; Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, 92 pp. (1977, August 20)

A conference was held at the Pennsylvania State University, University Park, PA, on July 6-7, 1977 as part of the National Waste Terminal Storage Program which is in progress to develop federal repositories for terminal storage and isolation of radioactive waste in geologic formations. This conference emphasized understanding the chemical and physical reactions to be expected in the vicinity of waste packages in the near term, during the period in which temperature and radiation effects are significant. This report is divided into three main sections: (1) Overview of OWI Waste/Rock Interaction Studies; (2) Reports of Individual Contributions (consisting in most instances of an abstract provided by the author plus additional material distilled from the notes of two Penn State personnel who served as recorders); and (3) Conclusions. A total of 28 presentations were given; those most applicable to this data base have been abstracted separately. (ERB)

Only abstracts are given for the presentations.

SITE SELECTION; STORAGE, GEOLOGIC; WASTE MANAGEMENT; WASTE STORAGE; WASTE DISPOSAL; WASTES, RADIOACTIVE; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; ROCKS; IGNEOUS ROCKS; SEDIMENTARY ROCKS; SHALES; BASALTS; GRANITES; EVAPORITES; SALT DEPOSITS; REPOSITORY; WASTE-ROCK INTERACTIONS

<443>
Peterson, J.A., A.J. Lobeit, C.W. Spencer, and R.A. Ullrich, University of Montana, Missoula, MT; El Paso Natural Gas Company; Texaco, Inc..

Sedimentary History and Economic Geology of San Juan Basin, New Mexico and Colorado.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 186-231), 253 pp. (1968, August)

The San Juan basin is a structural basin containing up to 4572 m (15000 ft) of sedimentary rocks ranging in age from Cambrian to Recent. Many sandstone reservoirs in the basin that would otherwise be suitable for waste disposal have been eliminated from consideration because they contain potable water, oil, gas, helium, coal, uranium, or other minerals. The extensive sedimentary history of the basin is directly related to the occurrence of economic deposits in the basin. Major reserves of petroleum are in Cretaceous and Pennsylvanian rocks, coal in Cretaceous, and uranium in Jurassic and Cretaceous. Much of the San Juan basin is considered potentially suitable for waste disposal, contingent upon such factors as depth, volume of waste, petroleum and mining activities, present and future groundwater needs, and other industrial and cultural considerations. The Chaco Slope probably is the most favorable area for waste disposal. The Entrada is the most widespread sandstone of potential waste-disposal value, especially in the central and southwestern parts of the basin. Other possible reservoirs include the Leadville Limestone (Mississippian dolomite) in the north, the Agua Zarca Sandstone of the Moenkopi Formation (Triassic) in the south and southeast, locally the Shinarump Conglomerate (Triassic) of the Chinle Formation, and possibly some Morrison Formation sandstones (Jurassic). Some units of the Mancos Shale (Cretaceous) may serve as receptors for waste-slurry grouts in artificial fractures, but the formation contains oil, coal, and water. The San Ysidro Member of the Yeso Formation (Permian), to the south of the San Juan basin could possible be used for storage of solid wastes in salt. (Auth) (DCW)

This manuscript was reprinted, with revisions, from American Association of Petroleum Geologists Bulletin 49(11):2076-2119 (1965). Though there is no specific mention of radioactive wastes in the paper, the principles and geologic conditions could also apply to radioactive wastes. Sixty-one references are provided, as well as 15 maps, 7 cross sections, 1 correlation chart, and 1 fence diagram.

WASTE DISPOSAL; WASTE STORAGE; WASTES, SOLID; WASTES, LIQUID; SEDIMENTARY ROCKS; SANDSTONES; SHALES; DOLOMITE; SALT DEPOSITS; GEOLOGY; ROCKS; GEOLOGIC FORMATIONS; BASINS, STRUCTURAL; GROUTING; HYDRAULIC FRACTURING; SITE SELECTION; FRACTURES

<444>
Piper, A.M.

Hypothetical Prototype Sites of Repositories for Radioactive Wastes: Flood Basalt.

Y/OWI/SUB-3745/6; 55 pp. (1975, April)

This report summarizes available geologic and hydrologic information in regard to the suitability of the flood basalt (plateau basalt) of the Pacific Northwest as the host medium of a hypothetical repository for high-level radioactive wastes. The particular setting considered is that of the Hanford Reservation at Richland, WA. The flood basalt sequence of the Pacific Northwest region is presently designated as the Columbia River Basalt Group; its lower part is the Picture Gorge Basalt Formation (Miocene), and the upper is the Yakima Basalt Formation (upper Miocene-lower Pliocene). The maximum known thickness of the flood basalt (over 10,655 ft) occurs 2 mi west of Hanford near the Wattlemate Hills. The stratigraphy and lithology of the layered extrusives and sedimentary interbeds is discussed, as are the tectonic and seismic features of the Pasco basin. The writer concludes that a waste repository deep in sound basalt beneath the Hanford Reservation would face only minimal prospective hazard from future tectonic or seismic activity. The available hydrologic literature indicates that, in general, the flood basalts tend toward isotropy in hydraulic conductivity. The piezometric surface for water in the flood basalt sequence in the vicinity of Hanford appears to be sharply trough-shaped along an axis approximately coinciding with the Columbia River, suggesting the water in the basalt is tributary to the river. Water

<444> CONT.
is withdrawn widely for use in public and domestic supplies in the region surrounding Hanford, and this withdrawal is likely to increase in future. Thus, any waste repository must guarantee that the waste is in such chemical and physical form, and is so managed in the repository vault, that no soluble toxic constituent can be dissolved by circulating groundwater and transported beyond the containment rate in concentrations exceeding the maximum permissible in public drinking water. Hydrologic characteristics of the basalt based on data from Hanford borehole ARH-DC-1 are given. These include: hydraulic conductivity 0.0016-6.8 ft/day (the high value is for a fracture zone); hydraulic storage coefficient 1.8 X 10(E-3)-1.8 X 10(E-6); effective porosity 1-10 vol% for basalt (high value for fracture zone) and 20 vol% for a sandy interbed. In the basalt to a depth of about 2700 ft the groundwater head is nearly constant at 168-173 ft below the surface, but at depths greater than about 3900 ft the head is 35-40 ft lower. Because the static head is about 15 ft above the mean level of the Columbia River, wastes placed in repository near bore hole ARH-DC-1 at a depth less than 2700 ft might be subject to a large downward hydraulic gradient (0.01 ft/ft) to the 3900-ft level; if this lower zone is extensive, and, as has been hypothesized, is an "artery" for groundwater discharge to the river (under a 3 X 10(E-4) ft/ft hydraulic gradient), a potentially hazardous situation might exist. The author concludes that there are critical uncertainties as to the feasibility of a radioactive waste repository in the flood basalt sequence beneath the Hanford Reservation; hydrologic data for the entire flood basalt province is meager, and further research in this area is needed. (Auth) (LKH)

Hydraulic Gradient; Hydraulic Head; Porosity; Hydraulic Conductivity; Storage Coefficient

BASALTS; IGNEOUS ROCKS; STORAGE, GEOLOGIC; WASTE STORAGE; WASTES, HIGH-LEVEL; STRATIGRAPHY; SITE SELECTION; GEOLOGY; SEDIMENTS; TECTONICS; GEOLOGIC STRUCTURES; ISOTROPY; HYDROLOGY; HYDRAULIC CONDUCTIVITY; RIVERS; GROUND WATER; EARTHQUAKES; POROSITY; STORAGE COEFFICIENT; HYDRAULIC GRADIENT; RADIATION HAZARDS; CONTAINMENT TRANSPORT; RADIONUCLIDE MIGRATION; FIELD STUDIES; THEORETICAL STUDIES

<445>
Radioactive Waste "Graveyards".

California Geology, p. 82. (1975, April)

For disposal of high-level wastes dry rock formations beneath arid deserts have been suggested. The solidified wastes would be placed at a depth of 10 to 100 feet beneath mesas and plateaus where the water is at depths of up to 2000 ft. The solidified wastes would be in the form of low solubility glasses and would then be placed at the bottom of 100 ft holes drilled into the thick unsaturated zone. The holes would then be filled with gravel and other backfill. The potential assets to this method are: aridity of the region, great depth of water table, solidified nature and low solubility of the wastes, and natural ion exchange properties of rocks; wastes will not be exhumed by erosion for a period of several thousand of years; suitable federally-owned lands make land acquisition unnecessary; and shallow burial of solidified radwastes permits retrieval. One of the main difficulties is in predicting erosion rates and climatic changes over thousands of years. Although much research needs to be done, deserts of the world constitute unused space with a potentially significant capacity to buffer the environment against degradation by radwastes. (HPV)

BEDROCK; BURIAL; CONTAINMENT; DEGRADATION; DISPOSAL SITE; GEOLOGIC STRATA; GROUND WATER; HYDROLOGY; MONITORING; RADIONUCLIDE MIGRATION; WASTE DISPOSAL; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE; WATER TABLE

<446>
Bauspott, L., and L. Bulloa, Lawrence Livermore Laboratory, Livermore, CA.

Heater Experiments in the Climax Stock, Nevada Test Site.

I/OBI/SUB-77/14268; CONF-770709; Waste-Rock Interactions, Proceedings of the National Waste Terminal Storage Program Conference, University Park, PA, July 6-7, 1977, pp. 57 [illegible] 72 pp. (1977, August 20)

In September, 1977, Lawrence Livermore Laboratory plans to operate three in situ heater experiments in an open drift 1000 ft below the surface in the Climax Stock at the Nevada Test Site. The stock is a quartz monzonite porphyry consisting of about 90% quartz and feldspar and minor amounts of other minerals, mainly biotite (7-8%). Chlorite (1-2%) is the only other mineral present in amounts greater than 1%. Quartz ranges from 15 to 30 vol%. The quartz monzonite is medium to coarse grained, but contains pink alkali feldspar crystals up to 5 cm in diameter. About 1 wt% of the rock is chemically combined water in biotite and other hydrous minerals. Another 1-2 wt% water is localized in small scale fractures and pores, and is thus free to migrate under thermal or pressure gradients. The experiment level would be above the regional water table and only a very small amount of water inflow occurs. Although the stock is bounded on the east and west by active faults and contains numerous faults, joints, and fractures, relatively unfractured and unfaulted areas are known which would be suitable for a nuclear waste simulation thermal experiment. Some of the faults and joints are filled with secondary minerals, and apparently many of the fractures are not interconnected, as measurements of background permeability in drill holes give values on the order of a few millidarcies to tenths of millidarcies. Three heater tests are envisioned. The first consists of a single heater surrounded by thermocouples at distances of 0.1 to 5 m. The heater will be 3 m X 0.019 m, and will generate 3 kw; the heater will be on for one month, off for one month, for a period of 8 months. The rock surface temperature in the heater hole is not expected to exceed 500-600 C and temperatures beyond 0.1 m into the rock should be below 400 C. The second experiment will be more completely instrumented to include measurement of permeability, rock displacement, stress/strain, and possibly acoustic emission. The heater will be operated continuously for 8 months. The third test will be a scale-up of the second, using multiple heaters energized for about one year. They will be arranged around a

<486> CONT.
pillar structure left in the room to obtain information on mine stability in the presence of multiple heaters. (Auth) (LEB)

Only the abstract is given.

HEAT SOURCES; FIELD STUDIES; STORAGE, GEOLOGIC; IGNEOUS ROCKS; GEOLOGIC STRUCTURES; PERMEABILITY; STRESS; STRAIN; THERMAL PROPERTIES; HEAT TRANSFER; REPOSITORY; WASTES, RADIOACTIVE; SITE SELECTION; WASTE-ROCK INTERACTIONS; MINERALS

<487>
Hasspott, L.D. (Ed.), Lawrence Livermore Laboratory, Livermore, CA.

Waste-Isolation Projects: Fiscal Year 1978.

UCRL-50050-78; 31 pp.(1979, January 12)

The report describes Lawrence Livermore Laboratory (LLL) activities during FY 1978 in support of the National Waste Terminal Storage Program. Current projects at LLL fall into three categories: (1) field testing, (2) laboratory rock mechanics measurements, and (3) laboratory studies of sorption and leaching. Field test activities conducted in the Climax granite at the Nevada Test Site included electrical heater tests, preparation for a spent-fuel-storage test, and planning for a series of rock mechanic tests. The heater tests determined the in situ thermal properties of Climax granite and its in situ permeability as a function of rock temperature. The two main laboratory rock mechanics projects involved (1) measurement of the permeability, electrical conductivity, and acoustic velocity of 15-cm-diam cores of granitic rocks over a range of confining pressure, pore (water) pressure, and deviatoric stress, and (2) measurement of rock thermal properties as a function of temperature and confining pressure in the presence of pore fluids to 770 K and 200 MPa. The leaching studies made use of an LLL-designed, single-pass leaching apparatus with three solutions, two leach temperatures, and three flow rates. The material evaluated was Np-Pu-doped simulated waste glass from Battelle Pacific Northwest Laboratories. The sorption studies involved standard static measurements of the equilibrium distribution coefficient (Kd) for various radionuclides on a variety of rocks, and flow-through-core studies of dynamic sorption. (Auth)

Temperature; Distribution Coefficient; Permeability; Leaching Rate; Thermal Expansion

FIELD STUDIES; LABORATORY STUDIES; ROCK MECHANICS; SORPTION; LEACHING; GRANITES; HEAT SOURCES; INSTRUMENTS; THERMAL PROPERTIES; PERMEABILITY; STRESS; STRAIN; GLASS; WASTES, RADIOACTIVE; DISTRIBUTION COEFFICIENT; SPENT FUELS; WASTES, HIGH-LEVEL; STORAGE, GEOLOGIC; SEISMOLOGY; IGNEOUS ROCKS; GROUND WATER; FRACTURES; REPOSITORY; GEOCHEMISTRY; PLUTONIUM; CESIUM; STRONTIUM; NEPTUNIUM 237; SITE SELECTION; PRESSURE EFFECTS

<488>
Reichert, R.G., Dornier System GmbH, Friedrichshafen, German Federal Republic.

Transportation of Radioactive Waste-Materials to the Sun.

CONF-721068; Nuclear Power and Propulsion Devices in Space, Proceedings of the 23rd Congress Symposium, Vienna, Austria, October 8-15, 1972. International Astronautical Federation, (38 pp.).(1972)

Initial results are given of a feasibility study of applying space technology for the removal of radioactive wastes to the sun. Solar transport of radioactive waste materials from future power plants appears to be feasible even with current space technology, although the cost per kWh would be rather high. The trajectory to be chosen depends on the time for the intended realization and the state-of-the-art (launch vehicle technology) available. In general four trajectories for solar impact are possible: direct transfer, bi-elliptical transfer, and Jupiter and Venus swing-by. The various methods and constraints for reaching the sun are discussed. Based on existing and/or planned launch vehicles the future payload-capabilities and logistics possibilities are considered. A mission-specific stage would have to be developed for this project making use of the energy of the waste for propulsion purposes and thereby avoiding larger cooling problems. The propulsion aspects for such a stage are treated parametrically, since no final choice can be made at this time. Safety considerations are briefly discussed. Cost and logistics aspects are given for the various possibilities in the near and longer-range future. Specific transport cost in the order of $3000/kg can be expected by 1985 (0.03 cent/kWh). This is also the order of magnitude expected for conventional storage. The launch frequency would be in the order of 100 to 300 launches per year. (RAF)

WASTES, RADIOACTIVE; TRANSPORTATION; SPACE TECHNOLOGY; SITE SELECTION; COST BENEFIT ANALYSIS; SPACE VEHICLES; PHYSICAL PROPERTIES; THERMAL PROPERTIES; SPACE DISPOSAL; STORAGE, EXTRATERRESTRIAL

<489>
Revelle, R., and H.B. Schaefer, University of California, Scripps Institution of Oceanography, La Jolla, CA; Inter-American Tropical Tuna Commission, La Jolla, CA.

Oceanic Research Needed for Safe Disposal of Radioactive Wastes at Sea.

A/CONF.15/P-2431; Peaceful Uses of Atomic Energy, Proceedings of the 2nd United Nations International Conference, Geneva, Switzerland, September 1-13, 1958. United Nations, Geneva, Switzerland, Vol. 18, (pp. 368-370).(1958)

An overview of the problems with radwaste disposal at sea is presented. Accidents to nuclear-powered ships, radioactive materials in coastal waters, and radwaste materials in the deep ocean waters are considered in relation to dispersal by water movement, deposition in the sediments, and uptake by marine organisms. The point is stressed that so very little is known about many of the physical, chemical, geological, and biological aspects of the sea that consideration of sea disposal is not at the time feasible. More research needs to be

<449>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<449> CONT.
done. (NEW)

SORPTION; ACCIDENTS; BIOSPHERE; CONTAMINATION; DISTRIBUTION; HYDRODYNAMICS; ENVIRONMENTAL EXPOSURE PATHWAY; OCEANOGRAPHY; RECOMMENDATIONS; SEA DISPOSAL; WASTE DISPOSAL; WASTES, RADIOACTIVE; REVIEWS

<450>
Richter, D., and W. Korner, Nuclear Safety and Radiation Protection Board, Berlin, German Democratic Republic.

Disposal of Radioactive Wastes Produced in Nuclear Installations in the German Democratic Republic.

CONF-760310; STI/PUB/433; IAEA-SM-207/44; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 2, (pp. 271-286), 424 pp. (1976)

As conventional energy sources have decreased in availability, the GDR has been developing its nuclear power production. Studies have been conducted to establish the best methods for disposal of radioactive wastes from the nuclear fuel cycle. Methods investigated included storage above ground, shallow burial, storage in salt mines and formations, and injection into porous bedrock. All types of radioactive waste were considered in the study and the various aspects compared from the standpoints of economics and nuclear safety. As a result, the solution chosen was the secondary usage of abandoned salt mines. Construction of such a repository is currently underway in the Bartensleben mine, located in the cleft salt diapir of the upper Aller Valley. Volume of the storage area is about 5 x 10(E+6) cu m. (LKN)

GEOLOGIC STRATA; SALT DEPOSITS; SITE EVALUATION; SITE SELECTION; STORAGE, GEOLOGIC; WASTE DISPOSAL; WASTE STORAGE; WASTES, RADIOACTIVE; COST BENEFIT ANALYSIS; STORAGE, ABOVEGROUND; REPOSITORY; WASTES, LOW-LEVEL; WASTES, HIGH-LEVEL, WASTES, INTERMEDIATE-LEVEL; DIAPIRS

<451>
Rockwell International, Atomics International Division, Department of Waste Isolation, Richland, WA.

Basalt Waste Isolation Program Annual Report.

RHO-ST-9; 56 pp. (1977, September 30)

In 1976, the Office of Waste Isolation, Union Carbide Corporation, funded a program to complete a study concerning the feasibility of storing nuclear defense waste in the deep basalts underlying the Hanford reservation. As part of the program, geologic, hydrologic, geochemical, and engineering studies have been conducted and all of these studies have emphasized the preliminary conclusion that the basalts appear to be a viable formation for a nuclear waste repository. This annual report provides information on these studies and on the work that was accomplished during the period from October 1976 through September 1977. (JST)

BASALTS; GEOCHEMISTRY; STORAGE, GEOLOGIC; HYDROLOGY; GEOLOGY; WASTE DISPOSAL; WASTE STORAGE; REPOSITORY; RECOMMENDATIONS; IGNEOUS ROCKS

<452>
Rogers, R.A., and L.J. Johnson, Los Alamos Scientific Laboratory, Environmental Studies Group, Los Alamos, NM.

Environmental Geology in Development of Nuclear Energy.

American Association of Petroleum Geologists Bulletin 59(5):921. (1975)

Establishment of nuclear waste management technology is a major need for full development of nuclear energy. Treatment and disposal of nuclear wastes are current research and development projects at such national laboratories as the Los Alamos Scientific Laboratory (LASL). Nuclear wastes with long half-lives must be isolated from the environment for periods of time which can be grasped best on a geologic time scale. Nuclear wastes with short half-lives must be isolated from the environment even though their period of isolation is on the order of a few tens of years. Discussion of proper site selection for geologic disposal considers whether the wastes have long or short half-lives and then considers the relation of such factors as climate, rock chemistry, hydrology, and structure to containment. The LASL has engaged in nuclear waste disposal since 1943. This experience forms the foundation for continuing efforts in solving nuclear waste problems. (Complete Text) (Auth)

WASTE MANAGEMENT; WASTE TREATMENT; WASTE DISPOSAL; STORAGE, GEOLOGIC; SITE SELECTION; CONTAINMENT; GEOLOGY; HYDROLOGY; GEOCHEMISTRY; WASTES, RADIOACTIVE

<453>
Sandia Laboratories, Albuquerque, NM.

Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Regional Geology.

SAND-78-1596; Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, D.W. Powers, et al. (Eds.), Vol. 1, Ch. 3, (pp. 3-1 - 3-112), 616 pp. (1978, December)

The regional geology and geologic history of southeastern New Mexico may be divided into 3 major intervals of about 0.5 billion years each. The period from about 1.5 to 1.0 billion years ago is represented by metamorphosed rocks of various sedimentary to igneous origins. From 1.0 to 0.5 billion years ago, no rock record was preserved; erosion is assumed to have been the dominant process during this time. From about 500 to 100 million years ago, the region shows a record of mostly marine sediments deposited in a basin that became progressively shallower. About 225 million years ago the marine environment became quite restricted, resulting in deposition of about 8000 ft of evaporites in which it is proposed to dispose of radioactive waste. The lack of rock

<453> CONT.
record in much of the region from about 100 million years to about 4 million years ago implies that erosion has been dominant. Pluvial sediments and caliche developed within the last 4 million years suggest some change in base level but considerable stability. Important structural features include basin and range-type faulting more than 70 mi west of the site and eastward tilting of the Delaware Basin, in which the site is located; both of these features are believed to be mid-Cenozoic in age. The Central Basin Platform to the east of the site has probably not been active since the Permian. (Auth) (LKM)

Physiographic and geologic maps, cross sections, stratigraphic correlations, tectonic and structure contour maps of the region are included at the end of the chapter.

Stratigraphic Unit Thickness

GEOLOGY; GEOLOGIC HISTORY; METAMORPHIC ROCKS; IGNEOUS ROCKS; SEDIMENTARY ROCKS; BASINS, STRUCTURAL; EVAPORITES; GEOLOGIC STRUCTURES; FAULTS; SITE SELECTION; STORAGE, GEOLOGIC; WIPP; FIELD STUDIES; REVIEWS; SALT DEPOSITS; WIPP; SITE EVALUATION

<454>
Sandia Laboratories, Albuquerque, NM.

Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Site Geology.

SAND-78-1596; Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, D.W. Powers, et al. (Eds.), Vol. 1, Ch. 4, (pp. 4-1 - 4-94), 616 pp. (1978, December)

WIPP site physiography and geomorphology shows the site to have been relatively stable for the last 500,000 years or more. The evaporites at the site are about 3500 ft thick, and include the Castile, Salado, and Rustler Formations from top to bottom. Subsurface structures within the evaporites include deformation 3 mi north of the site and possibly also 1 mi north. The structure 3 mi north involves deformation of the Castile and lower Salado; drilling (WIPP 11) showed the deformation to be upward bulging of salt, but no severe displacements, and no brine or gas were encountered. The structure 1 mi north appears to be smaller, and is of concern mainly for design of underground workings; it continues to be investigated. Within the site structure contours on evaporite horizons show areas with slight closure; the range of precision of data from industry sources and natural variability can probably account for these features though slight deformation or dissolution could also be invoked. No surficial faults are known at the WIPP site. The potential repository zones are in the Salado Formation at depths of 2730-2630 ft (remote handling) and 2176-2074 ft (contact handling). These beds were chosen on the combined bases of purity, depth, thickness, natural separation, and depth below the potash zone. (Auth) (LKM)

Maps illustrating site topography, location of drill holes, surficial geology, and isopach and structure contour maps of the site are included at the end of the chapter. Also included are selected well logs, a generalized site stratigraphic section, and a site geologic column.

Stratigraphic Unit Thickness

GEOLOGY; SITE SELECTION; WIPP; STORAGE, GEOLOGIC; EVAPORITES; SALT DEPOSITS; GEOLOGIC FORMATIONS; GEOLOGIC STRUCTURES; DEFORMATION; STRUCTURAL STABILITY; FIELD STUDIES; GEOLOGIC STRATA

<455>
Sandia Laboratories, Albuquerque, NM.

Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Seismology.

SAND-78-1596; Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, D.W. Powers, et al. (Eds.), Vol. 1, Ch. 5, (pp. 5-1 - 5-77), 616 pp. (1978, December)

Non-instrumental and regional instrumental studies of earthquakes prior to 1972 in southeastern New Mexico indicate that the most significant sources of earthquakes were the Central Basin Platform region near Kermit, Texas, and the area about 200 km or more west and southwest of the WIPP site (Rio Grande Rift Zone). The strongest earthquake reported to occur within 300 km was the intensity VIII (on the modified Mercalli scale) event of August 16, 1931, at Valentine, Texas, a distance of approximately 210 km. The closest shock (as of 1972) was a magnitude 2.8 event on July 26, 1972, about 40 km northwest of the site. The record from regional studies of events west and southwest of the WIPP site 200 km or more is consistent with the record of Quaternary faulting in that area. Instrumental studies near the WIPP site since 1974 and near Kermit, Texas, since late 1975 have recorded additional evidence of seismic activity for the site and region. The pattern obtained from near the site is similar to that from regional studies; about half of the located events in the data set occur on the Central Basin Platform while most of the rest occur to the west and southwest of the site in the Rio Grande Rift Zone. The data set also includes 3 events within about 40 km of the site since 1972. Two events have been assigned magnitudes of 2.8 and 3.6; the third event (in 1978) has only preliminary data available. Data for the Central Basin Platform from the Kermit array continue to show that location as the most active seismic area within 300 km of the site in terms of number of events. The largest earthquake known to occur in the Central Basin Platform had an estimated magnitude less than 3.25. Activity appears equally likely to occur anywhere along the Central Basin Platform structure without particular regard to small-scale structural features such as pre-Permian buried faults. The spatial and temporal coincidence of this seismicity with secondary petroleum recovery projects suggest a close relationship, but this has not yet been satisfactorily established. The lack of known Quaternary faults from the Central Basin Platform region is suggestive that large magnitude earthquakes are not occuring or have not occurred within the recent geologic past in the area. Analysis of regional and local seismic data indicate that the 1000 year acceleration is less than or equal to 0.06 g and the 10,000 year

<455>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<455> CONT.
acceleration is less than or equal to 0.1 g for all models tried. Probabilities at which higher accelerations occur depend almost exclusively on the assumptions made about the seismic potential of the immediate site area. (Auth) (LKM)

Regional seismicity and seismic station array maps, tables of earthquake dates and locations, plots of acceleration probability curves, and other data are provided at the end of the chapter.

SEISMOLOGY; EARTHQUAKES; SITE SELECTION; SITE EVALUATION; FAULTS; GEOLOGIC STRUCTURES; MONITORING; WIPP; FIELD STUDIES

<456>
Sandia Laboratories, Albuquerque, NM.

Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Hydrology.

SAND-78-1596; Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, D.W. Powers, et al. (Eds.), Vol. 2, Ch. 6, (pp. 6-1 - 6-66), 616 pp. (1978, December)

This chapter contains an assessment of surface and groundwater resources in and surrounding the proposed WIPP site and a status report of current hydrologic investigations in the local site area. The proposed site contains neither perennial streams nor surface water impoundment, and the water bearing strata above and below the evaporite section do not yield large quantities of water to wells. At its closest point, the Pecos River is 18 mi southwest of the site. Water bearing strata in the site area at stratigraphic horizons above the proposed repository include the Santa Rosa Sandstone and the Culebra and Magenta members of the Rustler Formation. Hydrologic units below the repository horizons include the Bell Canyon Formation of the Delaware Mountain Group. Ground water velocities vary with porosity, hydraulic conductivity, and head gradient, but are typically 0.3 ft/day in the Santa Rosa Sandstone, 0.5 ft/day in the Rustler Formation, and 0.0006 ft/day in the Bell Canyon Formation. Shallow wells in the area are generally used only for watering livestock and typically produce nonpotable water with total solute concentrations in excess of 3000 ppm. Natural potentiometric levels of fluids from rocks above and below the evaporites at the WIPP site (from the Rustler and Bell Canyon Formations, respectively) are comparable (about 2950-3050 ft above mean sea level). The head differential between upper and lower units varies within about 200 ft on the periphery of the site, the sign and magnitude of the differential depending upon precise geographic location. Gentle eastward tilting of the Delaware Basin has exposed the Salado Formation to near-surface waters and resulted in removal of a wedge-shaped mass of soluble salts between the Guadalupe Escarpment (30 mi due west of the site), where there remains no rock salt, and a point about 3 mi due west of the site center, where the Salado Formation is intact. The proposed site is in an area of the Delaware Basin that is free of regional deep dissolution, but localized shallow features are present in the vicinity. A shallow salt dissolution zone (called the "brine aquifer") occurs in Nash Draw at the contact between the Salado and Rustler Formations. This dissolution area is 2-10 mi wide and approximately 30 mi long. The brines flow southwesterly at a rate of about 0.2 ft/day and discharge into the Pecos River at Malaga Bend. The average rate of vertical dissolution is estimated to be 0.33-0.5 ft per 1000 years, and the average lateral dissolution rate has been estimated as 6-8 mi per million years. Climatological records show that mean annual precipitation at the site is about 12 in/yr. Aided by the low relative humidity (typically 36% during daylight hours) and high mean annual temperature (61 F), most of the precipitation returns to the atmosphere via evaporation and transpiration. Runoff is typically 0.1-0.2 in annually. (Auth) (LKM)

Hydrologic maps and cross-sections, well location maps, and precipitation and well records are included at the end of the chapter.

Hydraulic Velocity; Hydraulic Head; Mean Precipitation

HYDROLOGY; AQUIFERS; HYDRAULIC VELOCITY; HYDRAULIC HEAD; BASINS, STRUCTURAL; EVAPORITES; SALT DEPOSITS; EROSION; DISSOLUTION; BRINES; HYDRAULIC FLOW; PRECIPITATION, METEOROLOGICAL; HUMIDITY; TEMPERATURE; EVAPORATION; TRANSPIRATION; RUNOFF; WIPP; SITE SELECTION; SITE EVALUATION

<457>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations.

Y/OUI/TM-44; 837 pp. (1978, April)

The report concentrates on deep geologic isolation of wastes in bedded salt, granite, shale and basalt with emphasis on wastes from three fuel cycles: reprocessing wastes from uranium and plutonium recycling, reprocessing wastes from uranium-only recycling, and spent unreprocessed fuel with no recycling. The analyses presented are based on preconceptual repository designs, and are being submitted for input to the "Draft Generic Environmental Impact Statement on Commercial Waste Management" being prepared for the Department of Energy. The background of geologic isolation of radioactive wastes and a historical review of storage programs are given. Discussed in detail are: selection, description and geologic considerations of sites; descriptions of geologic formations; wastes and packaging; surface facilities; mining operations; safety; effluents; decommissioning; and geological alternative actions. Numerous references, tables, and figures are included. (CAJ)

Numerous artist conceptions and technical drawings of the various deep storage facilities are included. Tabular data also presented.

RECOMMENDATIONS; SITE SELECTION; WASTE STORAGE; WASTES, RADIOACTIVE; REPOSITORY; SALT DOMES; DISPOSAL SITE; DESIGN; SITE EVALUATION; STORAGE, GEOLOGIC; WASTE MANAGEMENT; HAZARD ANALYSIS; REPROCESSING; CONTAINMENT; MINING;

<857> CONT.
DECOMMISSIONING; EFFLUENTS; STANDARDS, FEDERAL; SPENT FUELS; ENVIRONMENTAL IMPACT STATEMENTS; MONITORING

<858>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Selection and Description of Generic Repository Sites.

Y/OWI/TM-88; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 2, (pp. 2-1 - 2-73), 837 pp. (1978, April)

This chapter deals with the general geologic considerations related to a site selection and describes 7 categories of potential geologic formations as salt deposits, crystalline rocks, argillaceous rocks, volcanic rock, and carbonate rocks. Some of the locations and geologic settings for these rock types are described. Four of the geologic media are considered for preconceptual design studies: bedded salt, granitic rocks, argillaceous rocks, and flood basalt. Generic site conditions, such as stratigraphy, hydrogeology, in-situ rock stresses, host rock properties, and environmental effects, are discussed. (CAJ)

Many tables, maps, technical drawings, stratigraphic charts, and references are included.

RECOMMENDATIONS; SITE SELECTION; WASTE STORAGE; WASTES, RADIOACTIVE; REPOSITORY; SALT DEPOSITS; STORAGE, GEOLOGIC; WASTE MANAGEMENT; ROCK MECHANICS; STRESS; GEOPHYSICAL SURVEYS; GEOLOGY; GEOLOGIC STRUCTURES; BASALTS; GEOLOGIC STRATA; GRANITES; CLAYS; SHALES; LIMESTONES

<859>
Smith, B.J.

Dissolution of Kansas Evaporites - The Radioactive Waste Disposal Problem.

Fourth Annual University of Missouri/Rolla-Missouri Department of Natural Resources Conference on Energy, J.D. Morgan (Ed.), Proceedings of a Conference, Rolla, MO, October 11-13, 1977, Vol. 4. (pp. 85-91). (1977)

The western two-thirds of Kansas is underlain by gently dipping evaporite-bearing rocks of Permian age. The most extensive salt unit is the Hutchinson salt in the Wellington Formation. Above the Permian evaporites are freshwater aquifers yielding water for domestic, industrial, and agricultural users, irrigation being the major use of water in the western counties. Waters in most of the Permian and lower strata are highly mineralized and of limited use. Under the Permian evaporites are the petroleum production zones which account for the 78.5 million barrels of crude oil and over 894 million MCF of natural gas produced in 1971. Consideration of the Permian salt deposits for radioactive waste disposal sites dates from 1955. Lyons, Kansas was selected as an experimental repository. The site was abandoned, however, because of unexplained losses of water in the repository area by a local salt extracting company and a long record of oil well drilling close to the site. The Lyons experiment focused attention on the problem of evaporite dissolution. Deteriorated casings or boreholes not adequately plugged allow water from above to enter the salt units and set up new hydrodynamic systems which adversely affect the groundwater supply. Petroleum drilling is not the only means of access to the salt. Others present in Kansas include salt solution extraction mining and natural factors such as faults. Many areas are mining the groundwater; over-pumpage of fresh water can allow lower, denser water to move upward when hydraulic communication is provided. There is disagreement on the extent of salt dissolution, and more research is needed to determine which combinations of possible mechanisms are responsible. With regard to radioactive waste storage in salt, short-term effects, mainly those of thermal expansion of stratigraphic units above and below the salt beds and possible changes in the hydrodynamics of the area, are of questionable safety in Kansas. Groundwater leaching is a possibility in short geologic time even in an originally watertight salt bed used for disposal. When past and present salt extraction and petroleum exploration and production in Kansas are added to the uncertainties of thermal effects the problem is compounded. The problem of evaporite dissolution has also become significant recently in western Kansas with the recognition of groundwater depletion. This has resulted in reassessment of state water planning and policies. (Auth) (LKH)

SITE SELECTION; EVAPORITES; SALT DEPOSITS; GROUND WATER; WASTE DISPOSAL; WASTE STORAGE; WASTE MANAGEMENT; STORAGE, GEOLOGIC; HYDROLOGY; GEOLOGY; BOREHOLES; WELLS; AQUIFERS; DISSOLUTION; BRINES; POLLUTION, WATER; REVIEWS; MINING; WATER TABLE; HYDRODYNAMICS; THERMAL PROPERTIES; WASTES, RADIOACTIVE; REPOSITORY

<860>
Smith, C.G., Jr., Louisiana State University, Baton Rouge, LA.

Geohydrologic Studies of Gulf Coast Interior Salt Domes.

CONF-770383; Y/OWI/SUB-77/16537; Salt Dissolution, Proceedings of the National Waste Terminal Storage Program Review Meeting, Austin, TX, March 29-30, 1977, (15 pp.), 38 pp. (1977, March)

Results are presented on an investigation on groundwater salinity around salt domes in northeast Texas salt dome basin. For a salt dome to be used as storage for high-level waste it must be free or nearly free of groundwater dissolution for a period of 250,000 years. Also the salt dome must be as tectonically stable as possible. Domes considered in this study were Steen, Mount Sylvan, Whitehouse, Bullard, Brooks, Keechi, and Palestine along with Grand Saline and Hainesville. The Eocene sands of the Wilcox Group have a potential hydrologic connection with salt domes and are likely to contain saline groundwater plumes. The general direction of groundwater movement in the sands is southeastward. Beneath the Wilcox Group, the Midway clays form an important

<460> CONT.

aquitard. The Midway clays lie unconformably on the Cretaceous system at a depth of about 5000 ft. The Woodbine sands are found in this system but at present do not show any salt dissolution. A salt dissolution rate was calculated from parameters derived from the study of electric logs. For the Bullard Dome the thickness removed is 75 ft/250,000 years; Hainesville Dome, 60 ft/250,000 yrs; Steen Dome, 50 ft/250,000 yrs; and Whitehouse Dome, 20 ft/250,000 yrs. The Palestine dome is isolated from the Wilcox sands by the uplifted and truncated Midway clay and deeper Cretaceous sediments. Highest salinities near the dome range from 1000 to 3000 ppm and this suggests that freshwater is recharged to the Wilcox at the outcrop areas around the dome. Along with other indications this dome is considered to be relatively stable hydrologically. The Bullard dome has a saline plume around it in the Wilcox sands. The origin of the plume is a result of actual solution of salt by waters of the Wilcox or vertical flow of deeper saline water up the flanks of the dome into the Wilcox sands. It is estimated that 75 ft/250,000 years salt is removed from the dome. This would make the dome unstable hydrologically. (NDV)

AQUIFERS, CONFINED; CONTAMINANT TRANSPORT; DISPOSAL SITE; SALT DOMES; GEOLOGIC FORMATIONS; LOGGING, ELECTRIC; HYDROLOGY; DISSOLUTION; CLAYS; GROUND WATER; PIEZOMETRIC SURFACE; SANDS; WASTE DISPOSAL; FIELD STUDIES

<461>

Smith, C.G., Jr., R.A. Barlow, H. Hoda, J.D. Martinez, J.E. Novik, and R.L. Thoms, Louisiana State University, Institute for Environmental Studies, Baton Rouge, LA.

Geohydrology.

Y/OWI/SUB-4112/37; An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 95-210), 475 pp.(1977, September 30)

Geohydrologic studies added new data for the two sites in the north Louisiana Salt Dome Basin: Vacherie and Rayburn's domes. Electric logs from Vacherie dome indicate a subsidence structure exists over the center of the dome. Apparently the cause of subsidence was extensive salt dissolution at the top of the dome during deposition of the top of the dome during deposition of the Cane River Formation, which was found at more than 300 ft below the surface. This finding may indicate the rate of salt dissolution for the 48 million years and the resulting cap rock formation during the same time. Water in the caprock and near the salt is saline and water in the Wilcox sands surrounding the dome is fresh to saline. The source of a saline plume in the Wilcox aquifer has been suggested. Drilling at Rayburn's dome indicates no salt dissolution and apparently no saline water exists in the Wilcox aquifer. Tests to determine the saline anomalies in the ground water around Mississippi salt domes met with small success due to paucity of data on Mississippi domes. Visits to various salt mines on the Gulf Coast indicated some were dry and some were wet. Contrasting hydrologic conditions of the mines and possible relationships between mine dryness (or wetness) and salt dome-mine configuration are presented. Features that deserve more study include oil, gas and water seeps, areas of large crystals, and recrystallization and blow-outs during salt mining. Petrographic examination of salt and caprock for two domes reveal no consistent trends in anhydrite concentration near the salt/caprock interface. Euhedral anhydrite crystals concentrate and align along salt crystal boundaries. More work is anticipated. (NDV) (CSF)

Chapter is well illustrated by cross-sections and structural maps. Appendices contain more data.

AQUIFERS; CAP ROCK; GEOLOGY; GROUND WATER; CONTAMINATION; HYDROLOGY; LOGGING, ELECTRIC; MINES; SALT DOMES; DISSOLUTION; ANHYDRITE; WATER; MINERALOGY; FIELD STUDIES

<462>

Snow, R.H., IIT Research Institute, Chicago, IL.

Growth and Shape of Salt Cavity--Computer Model and Calibration from Detroit Mine Experiment.

CONF-770383; Y/OWI/SUB-77/14537; Salt Dissolution, Proceedings of the National Waste Terminal Storage Program Review Meeting, Austin, TX, March 29-30, 1977, (1 p.), 38 pp.(1977, March)

A mathematical model was developed to simulate the process of salt dissolution within a cavity. The model will predict the size and shape of the cavity over a period of several decades, particularly with a slow feed rate. Bulk flow equations along with several empirical equations characterizing boundary flow at the wall are employed. For calibration Detroit Mine data was done systematically by the method of nonlinear regression. The brine concentrations were within 10% of those measured and the final cavity size and shape was within 5% of the measured. A prediction was done using a feed rate of 1 gal/hr and 1 gal/day over a period of 10,000 years. The results show the cavity grows to a wide-flaring shape and that significant growth occurs at the top layer. The prediction is believed to be basically correct. (NDV)

DISSOLUTION; CAVITIES; COMPUTER PROGRAMS; MODELS, MATHEMATICAL; EQUATIONS; NUMERICAL METHODS; SALT DEPOSITS; THEORETICAL STUDIES

<463>

Soussellier, Y., Commissariat a l'Energie Atomique, France.

Interaction Between Processing Method and Radioactive Waste Burial Site.

Transactions of the American Nuclear Society 20:691-692; CONF-750411; Nuclear Energy Maturity, Proceedings of the 1st European Nuclear Society Conference, Paris, France, April 21-25, 1975. European Nuclear Society and American Nuclear Society, (pp. 691-692).(1975)

Effective burial of radwaste should consider the interaction between the processing methods and burial site. A site may be chosen with the process in mind or the reverse. Generally, leaching rate is the criterion used to determine the suitability of processing method and burial site

<463> CONT.

conditions. However, the numerical data derived from leaching tests must be used only as guides because no data exist on leaching rates for long time periods (residence time for the waste in the ground). An adequate container adds a factor on safety and will tend to lower the leaching rate. Also, if retrieval of wastes is desired the container should facilitate recovery of the wastes from storage. For near surface land storage either the geologic nature of the soil (clay or shale) or geographic location (semi-desert) will provide the required retention of the radwaste. Coupled with a low leaching rate of the radwaste, additional guarantees of safety are ensured when considering exceptional weather conditions, avoiding continuous supervision, and taking into account small animals and microorganisms. Deep geologic storage in dry beds such as salt does not require the material to have a low leach rate. However, a problem may occur over long periods of time when the site might be forgotten. Special handling for wastes to be buried in wet deep geologic structures such as clay is not required, as long as the formation water is strictly stagnant at the site. Storage in mixed beds will require that the radwaste be processed so it has a low leach rate. Hence, the main safety factor is the burial site and a secondary factor is the leach rate. (NDV)

BURIAL; CLAYS; CONTAINERS; CONTAINMENT; DISPOSAL SITE; GEOLOGY; IMMOBILIZATION; LEACHING; SALT DEPOSITS; WASTE DISPOSAL; WASTES, RADIOACTIVE; WATER; WASTE TREATMENT; SITE SELECTION; ENVIRONMENT

<464>

Spitsyn, V.I., and V.D. Balukova, Institute of Physical Chemistry, Academy of Sciences of the USSR, Moscow, USSR.

The Scientific Basis for, and Experience With, Underground Storage of Liquid Radioactive Wastes in the USSR.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 237-248), 563 pp. (1979)

This paper is a summary of completed investigations described in the papers listed in the reference section, in which aspects of storage of liquid radioactive wastes in the USSR were verified in actual underground storage installations. The wastes, from research centers and industry, have been disposed of underground for 20 to 25 years with a surface area committed to ground installations of only 0.2 to 0.3 sq km. Factors to consider in predicting the suitability of a particular waste for a geological host medium include: regional hydrodynamics, direction and speed of groundwater, host rock structure and composition, and possible changes due to host rock/waste interactions such as chemical reactions, dissolution processes, or formation of precipitates that would alter the effective porosity of the rock or the composition of wastes. Some favorable characteristics of a host rock would be a location at depth, where water exchange is slow, with low permeability strata above and below, covering a large territory, and with sufficient sorption and filtration capacities. The physical and chemical properties related to waste/rock/groundwater compatibility are important in choosing a site or in pre-treatment of wastes, should be homogenized and in a pH range of 5 to 9, and components that would precipitate should be extracted or stabilized. A method of determining compatibility of wastes with water and each other, based on potentiometric titrations and nephelometric analysis, was worked out. Predictions of behavior of waste components underground involve the processes of sorption (which can be influenced by ionizing radiation and the presence of water or electrolytes), destruction, formation of complexes, hydrolysis (accelerated by temperature increases, and which leads to the formation of crystalline precipitates that prevent transfer of waste ions to a solid phase), and chemical interactions. Temperature and pressure play important roles in these processes. The specific accumulation of radionuclides in the geologic porous medium, and on their retention during precipitate formation. If such processes occur at high temperatures and pressures, then the principal mechanism for accumulation of radionuclides is sorption on rocks. The methodology for prediction of sorption and migration behavior of radionuclides, for evaluation of the heat field in the strata and for radiolysis processes and their influence on the storage of liquid wastes in geologic strata was confirmed in the studies discussed. Systems for waste control in strata have been developed, and it has been demonstrated that safe storage of radioactive wastes can be brought about. (Auth) (DCW)

Table 1 gives sorption capacities of rocks in Ci/Kg of rock (sandstones, limestone, aleurites, sand and clay) as a function of solution composition. There are 2 additional tables showing hydrolysis data, and there is one figure showing increase in maximum temperature in a stratum collecting wastes.

WASTE DISPOSAL; WASTE STORAGE; WASTES, RADIOACTIVE; WASTES, LIQUID; STORAGE, GEOLOGIC; HOST ROCKS; WASTE-ROCK INTERACTIONS; WASTES, COMMERCIAL; WASTE PROCESSING; WASTE TREATMENT; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; GEOLOGY; ROCKS; GEOLOGIC STRATA; SITE SELECTION; SORPTION; PRESSURE EFFECTS; TEMPERATURE; RADIATION EFFECTS; RADIOLYSIS; HYDRODYNAMICS; COMPLEXES; RADIOACTIVE DECAY; DECAY HEAT; RADIONUCLIDES

<465>

Stanley, M.G., and D.W. Kapiasek, Interstate Electronics Corporation, Environmental Engineering Division, Anaheim, CA.

A Bibliography on Ocean Waste Disposal.

IEC-8060C0417; 143 pp. (1976, September)

This bibliography was prepared for the USEPA Office of Water Program Operations, to assist in the development of criteria for the control of sea disposal of wastes. It lists recent publications in the areas of general ocean disposal, non-point source oceanic disposal of radioactive wastes (51 documents), and at-sea incineration. More than 600 documents are listed, and 82 of them are abstracted. Titles concerning radioactive waste disposal included in the annotated abstracts are: A Survey of the

<465>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<465> CONT.
Farallon Islands Radioactive Waste Disposal Site for the U.S. Environmental Protection Agency, and Radioactive Waste Disposal Activities in the Vicinity of the Farallon Islands, for the U.S. Environmental Protection Agency. A list of identifiers for computerized accession is included. (LKH)

BIBLIOGRAPHIES; SEA DISPOSAL; WASTES, RADIOACTIVE; WASTES, NUCLEARRADIOACTIVE; INCINERATION; DISPOSAL SITE; WASTE DISPOSAL

<466>
Stephansson, O., and L. (Translator) Murphy, Kernbrennstoffsicherheit, Stockholm, Sweden.

Deformation of a Jointed Rock Mass.

ORNL/tr-4567; 85 pp.; KTS-TR-29; 57 pp.(1977, September 20)

Simple shear is a common tectonics in nature. Stress distribution, deformations and failure of a rock mass in direct shear loading is examined. If the strength of the rock mass is in the form of the Coulomb-Navier criterion one set of tensile fractures and two sets of shear fractures arise. The sets of joints predicted by the theory have been used in the mathematical modelling. Two computer models have been applied to the analysis of the jointed rock mass; the Discrete Block Method (DBM) and the Finite Element Method (FEM). DBM simulates the motion of a set of discrete blocks each one having the possibility of unlimited translation and rotation. Shear forces arise due to friction, damping forces are introduced to dissipate kinetic energy and contact forces and displacements are determined. The blocky rock models have been tested. One model simulates the deformation in a shear box. The two others simulate a regular and unregular jointed rock mass subjected to direct shear. Displacements and total deformation are more flexible as the number of blocks increases for one and the same model. Finite element analysis of a jointed rock mass is presented. The rigid blocks are assumed to be of linear elastic material and the joints are simulated by means of joint elements. Square blocks of dimension 4 x 4 km are cut by two sets of joints and the block is sheared an angle of 0.5 degrees. In models with a large joint spacing the rigid blocks reach contact at few points under high stresses. Models with small joint spacing show a more gentle and flexible mode of deformation and the stresses are much lower. Models loaded along the edges behave similar. Results of the DBM and FEM analysis in this study indicate that a suitable rock mass for repository of radioactive waste should be moderately jointed (about 1 joint /sq m) and surrounded by shear zones of the first order. This allows for a gentle and flexible deformation under tectonic stresses and prevents the development of large cross-cutting failures in the repository area. (Auth)

ROCK MECHANICS; STRESS; DEFORMATION; JOINTS; GEOLOGIC STRUCTURES; GEOLOGY; PHYSICAL PROPERTIES; ROCKS; STRAIN; MODELS; MODELS, MATHEMATICAL; COMPUTER PROGRAMS; THEORETICAL STUDIES; REPOSITORY; STORAGE, GEOLOGIC; DESIGN; WASTES, RADIOACTIVE

<467>
Stone, J.A., Savannah River Laboratory, Aiken, SC.

Evaluation of Concrete as a Matrix for Solidification of Savannah River Plant Waste.

DP-1440; 116 pp.(1977, June)

The properties of concrete as a matrix for solidification of Savannah River Plant (SRP) high-level radioactive wastes were studied. Concrete specimens were prepared and evaluated with both simulated and actual SRP waste sludges; properties were found adequate for fixation of SRP wastes. Procedures were developed for preparation of simulated sludges and concrete-sludge castings. Effects of cement type, simulated sludge type, sludge loading and water content on concrete formulations were tested in a factorial experiment. Compressive strength, leachability of strontium and plutonium, thermal stability, and radiation stability were measured for each formulation. From these studies, high-alumina cement and a portland-pozzolanic cement were selected for additional tests. Incorporation of cesium-loaded zeolite into cement-sludge mixtures had no adverse effects on mechanical or chemical properties of waste forms. Effects of heating concrete-sludge castings were investigated; thermal conductivity and DTA-TGA-EGA data are reported. Formulations of actual SRP waste sludges in concrete were prepared and tested for compressive strength; for leachability of Sr 90, Cs 137, and alpha emitters; and for long-term thermal stability. Concrete waste forms were generated which have compressive strengths of 2000-3000 psi for 40% sludge compositions, Sr 90 leachabilities of 10(E-6)-10(E-4) g/sq cm d, alpha emitter leachabilities of 10(E-8)-10(E-5) g/sq cm d, and long-term thermal stability up to 400 degrees C. Cs 137 leachabilities were on the order of 10(E-3)-10(E-1) g/sq cm d, which may necessitate washing of sludges to remove free Cs or incorporation of zeolite into the cement. It was found that high steam pressures were generated when the concrete forms were heated to 100-500 degrees C in closed containers, possibly requiring degassing and/or design of containers to accomodate high pressures and temperatures. (Auth) (LKH)

Leaching Rate; Thermal Conductivity

Cs 137; Sr 90; Pu

SOLIDIFICATION; CONCRETES; WASTES, HIGH-LEVEL; SLUDGES; LEACHING; STRONTIUM; PLUTONIUM; CEMENTS; CESIUM; ZEOLITES; CHEMICAL PROPERTIES; WASTE MANAGEMENT; WASTES, LIQUID; LABORATORY STUDIES

<468>
Stone, J.A., J.A. Kelley, and T.S. McMillan, Savannah River Laboratory, Separations Chemistry Division, Aiken, SC.

Sampling and Analyses of Savannah River Plant High-level Waste Sludges.

DP-1399; 50 pp.(1976, August)

Twelve 3-liter samples of high-heat waste sludges were collected from four Savannah

<968> CONT.
River Plant waste tanks with a hydraulically operated sample collector of unique design. Ten of these samples were processed in Savannah River Laboratory shielded cell facilities, yielding 5.3 kg of washed, dried sludge products for waste solidification studies. After initial drying, each batch was washed by settling and decantation to remove the bulk of soluble salts and then was redried. Additional washes were by filtration, followed by final drying. Conclusions from analyses of samples taken during the processing steps were: (a) the raw sludges contained approximately 80 wt % soluble salts, most of which were removed by the washes; (b) Sr 90 and Pu 238, Pu 239 remained in the sludges, but most of the Cs 137 was removed by washing; (c) small amounts of sodium, sulfate, and Cs 137 remained in the sludges after thorough washing; (d) no significant differences were found in sludge samples taken from different risers of one waste tank. Chemical and radiometric compositions of the sludge product from each tank were determined. The sludges had diverse compositions, but iron, manganese, aluminum, and uranium were principal elements in each sludge. Sr 90 was the predominant radionuclide in each sludge product. (Auth)

Chemical composition and residual leachable components of sludge, and specific activity of radionuclides are given in tabular form. The engineering design of the sludge collector is described in the appendix.

SLUDGES; SAMPLES; STRONTIUM 90; PLUTONIUM 239; PLUTONIUM 238; CESIUM 137; WASTES, HIGH-LEVEL; WASTE STORAGE; TANKS; SALTS; WASHING; SEPARATION PROCESSES; DRYING; LABORATORY STUDIES; SODIUM COMPOUNDS; CHEMICAL ANALYSIS; RADIOCHEMISTRY; IRON; MANGANESE; ALUMINUM; URANIUM

<969>
Swan, G., University of Lulea, Division of Rock Mechanics, Lulea, Sweden.

The Mechanical Properties of Stripa Granite.

LBL-7078; SAC-03; 25 pp.(1978, August 8)

The mechanical properties of Stripa granite are presented as determined from small (laboratory size), oven-dried specimens. The properties determined include Young's modulus, Poisson's ratio, uniaxial compressive fracture stress, and the expansion coefficient, all as a function of temperature. Samples were collected from boreholes UZE-1, DBH-2, Bh V1, Bh S1, hand specimens, and from oriented block B. Mean values for fracture stress at temperatures of 20, 50, 100, 150, and 190 C were (in MPa) 207.6, 208.2, 221.3, 205.5, and 148.0, respectively; Young's modulus values for the same temperatures were 69.4, 71.2, 62.4, 57.2, and 50.0 MPa, respectively. Mean Poisson's ratio values were 0.21, 0.21, 0.20, 0.16, and 0.13, respectively. Modal composition of the Stripa granite is 43.6% quartz, 12.0% potash feldspar, 39.2% plagioclase feldspar, 2.3% muscovite, and 3-2% chlorite. Experimentally determined volumetric thermal expansion coefficient is 33.4 X 10(E-6)/C. In addition, the Brazilian tensile fracture stress, residual shear strength as a function of a normal stress, and the rock's anisotropy ratios are presented. Finally ultrasonic determinations at 1 MHz of the rock's dilatational wave velocity are given and the deduced Young's modulus is compared with the static value for room temperature. The Stripa granite from the site locations is a relatively coarse-grained material which is laboratory tests exhibits linearly elastic behavior. In comparison with other granites both its Young's modulus and its compressive fracture stress are high. This may be accounted for by its high quartz content. The temperature dependency of its elastic properties within the range 25-200 C is similar to that of a granite reported elsewhere. The large-scale properties of Stripa granite will to a great extent be determined by its strongly jointed nature. This may be inferred with even greater certainty where chlorite-filled joints exist as a result of retrograde metamorphism. (Auth)(LBL)

Young's Modulus; Poisson's Ratio

ROCK MECHANICS; STRESS; THERMAL PROPERTIES; PHYSICAL PROPERTIES; LABORATORY STUDIES; GRANITES; JOINTS; REPOSITORY; STORAGE, GEOLOGIC; FRACTURES; MINERALS; IGNEOUS ROCKS; PLUTONS; ANISOTROPY

<970>
Tamanagi, H.V., J.E. Gale, and B.V. Sanford, Atomic Energy of Canada Limited, Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, Canada; Geological Survey of Canada, Terrain Sciences Division, Ottawa, Ontario, Canada; Geological Survey of Canada, Regional and Economic Geology Division, Ottawa, Ontario, Canada.

Underground Disposal of Canada's Nuclear Waste.

Geoscience Canada 4(2):71-76.(1977, June)

An outline is given of Canadian programs in progress and planned to insure the long-term underground isolation of radioactive wastes. A program to determine the feasibility and safety of geologic disposal by constructing a test repository in a suitable rock formation was initiated in 1975. The preliminary conceptual design for a repository which would contain all radioactive waste produced in Canada until 2050 has been completed. The test repository is expected to be completed by about 1986. A broad spectrum of geoscientific involvement is necessary to bring the program to a successful conclusion. A fundamental guideline which has been adopted in Canada is that concepts which rely on future development of new technology shall not be considered at this time. Host rocks under investigation for geologic disposal are plutonic crystalline rocks, salt deposits, and serpentinized ultramafic rocks. Brief descriptions are given of rock types and their characteristics and of site selection studies. (MAF)

STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; REPOSITORY; ROCKS; WASTE VOLUME; THERMAL PROPERTIES; WASTE DISPOSAL; SALT DEPOSITS; JOINTS; FRACTURES; LEACHING; GROUND WATER; PERMEABILITY; HALITE; HAZARD ANALYSIS; AQUIFERS; IGNEOUS ROCKS; GRANITES; MAPS; OUTCROPS; SITE SELECTION; DRILLING; POTASH; SURFACE WATERS; GLACIERS; GEOPHYSICAL SURVEYS; MONITORING; METAMORPHIC ROCKS; FIELD STUDIES

<471>

GEOLOGY, HYDROLOGY AND SITE RESOURCES

<471>
The Disposal of Radioactive Waste on Land.

NAS-NRC Publication No. 519; 142 pp. (1957, September)

Proceedings of a radwaste disposal conference are presented. The committee on shallow disposal considered four general geologic categories: surface excavation, infiltration into shallow permeable beds, natural and artificial excavations, and artificial solution cavities. For all the discussions 1,000 gallons of waste/day for a 1 megawatt reactor criterion is used. Quarries in granite and other crystalline rocks are feasible only if the wastes can be permanently immobilized. It is felt that grouting and self-sealing are probably unreliable. Excavations in permeable non-crystalline rocks such as sandstone, limestone, and coal are not worth considering unless the waste can be permanently immobilized in a solid form because it is virtually impossible to seal them. Any leakage might contaminate ground water supplies. Disposal of wastes in shale or clay pits might be a possibility because the material is self-sealing and adsorptive. However, it is too risky in the present state of knowledge. Disposal of radwastes in shallow aquifers below the water table is not recommended and may only be possible if further tests with clays indicate good retention of the waste. Using beds above the water table might be worth trying in isolated desert areas. However, more research is needed. Natural caverns in the zone of water are totally unsuitable for disposal of liquid wastes due to their permeability. Shallow mines have the same problem as natural caverns. Deep mines may be suitable for storage of dry wastes in containers. Heat would be a problem though. Special excavations such as in solid rock have the advantages of deep mines but would be expensive. The concept is worth investigating. Salt cavities are the most promising method of disposal for liquid radwastes. However, salt does flow under pressure and seismic activity may fracture the salt allowing leakage. Discussion is presented on the use of cribs and pits at Hanford and Oak Ridge. A portion of the proceedings presents high-level wastes disposal problems. (NDV)

ADSORPTION; BEDROCK; CONTAINERS; CONTAINER INTEGRITY; CONTAINMENT; DISPOSAL SITE; GEOLOGY; GROUND WATER; HYDROLOGY; IMMOBILIZATION; RECOMMENDATIONS; SALT DEPOSITS; CLAYS; SITE SELECTION; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL

<472>
Thom, R.L., T.A. Manning, Jr., L.K. Paille, R.W. Perritt, R.W. Gable, J.K. Whisman, R.T. Toney, R.C. Farmer, and S.L. Oden, Louisiana State University, Institute for Environmental Studies, Baton Rouge, LA.

Geomechanics.

T/OWI/SUB-8112/37; An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 21-98), 475 pp. (1977, September 30)

The major objective of the geomechanics effort in this investigation is to determine current rates of any possible ongoing salt dome movement. At present, bounds on vertical domal movement may be estimated and more precise bounds will be established as additional data becomes available. Measurements of the crustal movements by laser ranging, tiltmeters, and precise releveling over Vacherie and Bayhern's domes will provide the needed data. A recording seismograph will be added to the instrumentation over Vacherie dome. Numerical modelling of the Vacherie salt dome is to the point where realistic 3-space and plane strain models may be employed. Modelling of the natural temperature field around the salt dome is also being persued. Description of the soils around the tiltmeters as well as the determination of the mechanical properties of the soils has been performed. A borehole closure study for ascertaining rock salt properties has been planned. A 3-space finite difference program is being adapted to model saline plumes around salt domes. This will be useful in determining rates of salt dissolution and associated hydrologic stability of salt domes. A microseismic monitoring system for detection of acoustic emissions as a result of dome growth or dissolution is in place. The recommendation of the committee is to carry on with the programs already in progress. (NDV)

Appendices contain additional data.

COMPUTER PROGRAMS; DISPOSAL SITE; EQUATIONS; FIELD STUDIES; GEOLOGY; GEOPHYSICAL SURVEYS; GROUND WATER; HYDROLOGY; MODELS, MATHEMATICAL; MONITORING; WASTE-ROCK INTERACTIONS; SALT DOMES; SEISMOLOGY; WASTE DISPOSAL; STRESS; SOILS

<473>
Tomlinson, H., Atomic Energy of Canada, Limited, Whiteshell Nuclear Research Establishment, Chemistry and Materials Science Division, Pinawa, Manitoba, Canada.

Canada Keeps its Disposal Options Open.

Nuclear Engineering International 23(266):43-45. (1978, January)

Various alternatives to water storage have been examined and a development program is in progress to establish a pilot facility for the deep underground disposal of CANDU wastes. The overall program is aimed at developing final disposal capabilities while sustaining various storage and disposal options until the social, economic, health and environmental implications of nuclear power have been adequately resolved. Immediate storage at CANDU reactor sites is in water-filled pools. The design life for the pools is 30 years. A comparative study and cost analysis was completed in 1976 for a variety of storage options. The main options considered were: Wet storage (a) pools at reactor site and b) pools at a central site) and dry storage (c) air-convection vault, d) air-conduction vault, e) concrete canisters, and f) underground salt bed). Wet storage in pools, options a and b, is an established technology and current costs are known. Dry storage in vaults, options c and d, is considered to be an available but not demonstrated technology. Dry storage concrete canisters are an available option for which practical demonstration is in progress. Dry storage in underground salt beds, option f, requires further study before

<073> CONT.
it can be considered an available option. (Auth) (RAF)

WASTE DISPOSAL; WASTE MANAGEMENT; STORAGE, GEOLOGIC; COST BENEFIT ANALYSIS; STORAGE, ABOVEGROUND; CONTAINERS; SALT DEPOSITS; RETRIEVABILITY; ROCKS; WASTES, RADIOACTIVE; DESIGN; SPENT FUELS; REACTORS, CANDU; THEORETICAL STUDIES

<074>
Tepp, S.V., Savannah River Laboratory, Aiken, SC.

Summary - Alternatives for Long-Term Management of Defense High-Level Radioactive Waste.

Policy and Technical Issues Pertinent to the Development of Environmental Protection Criteria for Radioactive Wastes, Proceedings of an EPA Workshop, Albuquerque, NM, April 12-14, 1977, (28 pp.). (1977)

This paper summarizes the alternatives under consideration for the long-term management of defense high-level waste at SRP, taking into account relative costs, risks, and uncertainties; it does not present any recommendations. Twenty-three alternative plans are described, falling into four main classes: (1) convert waste to a leach-resistant form, such as canned glass cylinders, and ship to a Federal repository; (2) convert waste to a leach-resistant form and store in a surface facility on-site; (3) reconstitute waste as a slurry and dispose of it in a cavern under the site; (4) continue present storage of waste in tanks as salt cakes and sludge. Risks are summarized, taking into account routine radiation releases, major process incidents, natural disasters, sabotage, airplane crashes, and site abandonment. Costs (in 1976 dollars) and risks involved in the four main alternative classes are summarized as follows: alternative (1), budgetary cost $2.7 billion, population dose risk 0.90 man-rem, total effective cost $2.7 billion, incremental cost/risk $81,000/man-rem; alternative (2), budgetary cost $2.9 billion, population dose risk 0.22 man-rem, total effective cost $2.9 billion, incremental cost/risk $40,000/man-rem; alternative (3), budgetary cost $0.18 billion, population dose risk 62 man-rem, total effective cost $0.28 billion, incremental cost/risk = base; alternative (4), budgetary cost $0.28 billion, population dose risk 1.4 man-rem, total effective cost $0.28 billion, incremental cost/risk $1000/man-rem. (LKH)

COST BENEFIT ANALYSIS; WASTES, HIGH-LEVEL; WASTE MANAGEMENT; METHODS; GLASS; STORAGE, GEOLOGIC; STORAGE, ABOVEGROUND; TANKS; ACCIDENTS; WASTE STORAGE

<075>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Identification of Alternatives for Idaho National Engineering Laboratory Transuranic Waste Management.

DOE/ET-0001; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 7, (pp. 7-1 - 7-23), 586 pp. (1979, February)

This section reviews communications between state and federal officials concerning the Idaho National Engineering Laboratory (INEL) TRU waste and discusses earlier and ongoing studies of INEL waste management. Conclusions of two studies by the Committee on Radioactive Waste Management of the National Research Council with possible implications at the INEL are summarized. The studies are: shallow land burial of low-level radioactive wastes, and management of radioactive wastes at Hanford. Guidelines and terminology formulated for the present study are given. Six waste management alternatives are briefly described. The alternatives derive from three different premises: 1) a Federal Repository will be available as scheduled in 1985; 2) a Federal Repository will be available in the year 2005; and 3) a Federal Repository will not be available and the waste must remain in INEL. (RAF)

WASTES, TRANSURANIC; WASTE MANAGEMENT; WASTES, LOW-LEVEL; BURIAL; REPOSITORY; WASTE PROCESSING; TRANSPORTATION; RETRIEVABILITY; SOILS; WASTE VOLUME; DENSITY; HAZARD ANALYSIS; COST BENEFIT ANALYSIS

<076>
Van de Voorde, N., J. Claes, G. Dumont, H. Spriet, and K. Peeters, Centre D'Etude De L'Energie Nucleaire - Studiecentrum voor Kernenergie, Belgium.

Radioactive Waste Processing.

BLG-520/77; Annual Scientific Report 1976, (pp. 5-12 - 8-14), 168 pp. (1977)

Radioactive waste processing by SCK/CEN Radioactive Waste Disposal Department operated by Belgonucleaire in 1976 is summarized. 12,500 cu m of warm liquid waste was released to the intermediate-level liquid waste treatment plant; 59,103 kg of dry filtered sludge concentrates were bituminized to 63.5 cu m and conditioned in 802 standard size steel drums or in concrete-lined drums; 226,700 cu m of treated liquid waste (22,626 mCi) was released to the Mathe River, representing 41.9% of the total weighted activity release permitted to the river annually. 613 cu m of solid low-level combustible waste was incinerated to 22 cu m of ash; 440 cu m of incombustible and alpha-contaminated solid wastes were baled and conditioned (reduction factor 2.7); 151 cu m of non-combustible, incompressible waste was bituminized or incorporated in concrete and conditioned in steel drums; 6.5 cu m of medium and high-level waste was stored at Esterio Storage Site I; total waste stored equals 120.7 Ci alpha, beta, gamma, 803.8 Ci tritium, 238.7 g Pu, and 7.6 kg U. 2,283 metric tons of insolubilized solid waste with a gross alpha, beta, gamma activity of 706 Ci and 1,000 Ci tritium was dumped at a selected site 4500 m deep in the Atlantic Ocean. (LKH)

Waste Volume

H 3; Pu; U

WASTES, LIQUID; WASTES, RADIOACTIVE; WASTE TREATMENT; WASTE MANAGEMENT; BITUMINIZATION; SEA DISPOSAL; INCINERATION; IMMOBILIZATION; TRITIUM; URANIUM; PLUTONIUM; WASTES, SOLID; WASTE VOLUME;

<076> CONT.
DRUMS; SLUDGES

<077>
Vilks, G., Atlantic Geoscience Centre, Bedford Institute of Oceanography, Dartmouth, Nova Scotia, Canada.

The Disposal of High-Level Nuclear Waste in the Oceans.

Geoscience Canada 3(4):295-298. (1976, November)

An international workshop on the feasibility of sea disposal of high-level waste was held at Woods Hole, MA, February 16-20, 1976. Areas of discussion covered current knowledge on disposal technology, what needs to be known and with what certainty, potential criteria for disposal decisions, and programs needed. Most criteria discussed at the workshop point to the centers of mid-latitude gyres as the safest disposal sites; they are regions of relative geologic stability, low organic productivity, and essentially homogeneous sediments. It is suggested that future studies include in situ experiments on the transfer of organic matter through marine ecosystems, studies of chemical and physical responses of sediments to heat and radioactivity, of interstitial fluid behavior and ion exchange in the sediments, mass fluidization by heat, and viscoelastic properties of the sediments. Sandia Laboratories and other institutions in the US have been conducting research to identify barriers to radionuclide migration in solidified waste, walls of containers, basement rock, sediment, the benthic boundary layer, and seawater. Waste emplacement schemes considered include use of a free-falling or winch-controlled penetrometer in unconsolidated sediments, and construction of boreholes accomodating up to 800 waste canisters and sealed with grout in bedrock. In the UK, British Nuclear Fuels, Ltd. is working with problems of solidification and emplacement of high-level waste, and has proposed dumping it at sea in containers within a few years of release from the reactor, letting it cool by contact with seawater; archaeological artifacts are being studied to identify materials with long-term resistance to seawater for use in containers. A model developed by the National Radiological Protection Board predicts that by 2000 AD there will be about 400,000 tons of waste worldwide, yielding 10(E+9) Ci of actinides and 10(E+10) Ci of fission products. Assuming all waste is vitrified and sea disposed, leaching immediately at 10(E-5) g/sq m/day, calculated ratios of the estimated annual intake of each radionuclide by man vs. the maximum permissible intake should be in the range of 10(E-2) - 3 x 10(E-6). Japan is conducting an intensive feasibility study on the sea disposal of high-level waste, particularly liquid waste-impregnated concrete in drums. A brief statement of Canadian waste policy and of factors precluding the disposal of wastes in the Arctic Ocean or Canadian Arctic Archipelago was presented. Canadian participation in sea disposal research is strongly advocated. (LKH)

SEA DISPOSAL; WASTES, HIGH-LEVEL; SEDIMENTS; SEAWATER; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; RADIATION EFFECTS; MODELS; METHODS; WASTE STORAGE; WASTE MANAGEMENT; CONTAINMENT; PREDICTIONS; DOSE RATE; RECOMMENDATIONS

<078>
Walters, R.F., Walters Drilling Company, Wichita, KS.

Salt Dissolution in Oil and Gas Test Holes in Kansas.

I/OWI/SWD-77/16537; CONF-770303; Salt Dissolution, Proceedings of the National Waste Terminal Storage Program Review Meeting, Austin, TX, March 29-30, 1977, (2 pp.), 30 pp. (1977, March)

Thirteen subsidence areas in the Hutchinson Salt Member of the Permian Wellington Formation were investigated. In general the Member is 350 ft thick with 20% of the thickness comprised of shale and anhydrite interbeds in the rock salt. The Member occurs from depths ranging from 200 to over 2500 ft. Only on the updip eastern edge has erosion by solution occurred. This was a result of access to the Pleistocene and present water tables. In 1910 subsidence within the salt works of the Joy Morton Salt Company affected an area 150 ft in diameter and 15 ft vertically. Vertical subsidence of a few inches with a diameter of 600 ft resulted from the operation of Carey Salt Company's brine wells in 1925. In 1974 a sinkhole 360 ft in diameter formed. The volume of the sinkhole was 90,000 cu yd with a water level 21.5 ft below ground surface. The most intensely studied subsidence areas associated with oil operation are the Crawford and Hitt Sinks in the Gorham Oilfield. Subsidence during a 20 year period was 26 and 17 ft, respectively. In 1959 subsidence occurred quickly at the Planning Sinkhole creating a circular pit 300 ft in diameter and 85 ft deep. The Hutchinson salt was penetrated from 975 to 1275 ft by an abandoned salt water disposal well near the Planning Sinkhole. Evidence shows that solution of salt during modern rapid rotary drilling using fresh water results in borehole enlargement about 3 times the diameter of the drilled hole. Early rotary drilling resulted in enlargement of about 5 times. Neither amount is large enough to cause surface subsidence. Ordinarily, no salt dissolution occurs after drilling ceases. From this extensive search land subsidence areas in central Kansas associated with rock salt dissolution show that such subsidence areas attributable to man's activities are rare and unusual features. Oil-related subsidence areas occur in a ratio of 1:10,000 holes in the Hutchinson Salt. (NDV)

Stratigraphic Unit Thickness

ANHYDRITE; BEDROCK; DISSOLUTION; GEOLOGIC STRATA; MINING; PITS; SALT DEPOSITS; SHALES; SINKHOLES; WELLS; BRINES; DRILLING; WATER TABLE; FIELD STUDIES

<079>
Walters, R.F., Walters Drilling Company, Wichita, KS.

Salt Dissolution in Oil and Gas Test Holes in Central Kansas, Part 1: Salt Beds in the Subsurface in Russell, Lincoln, Ellsworth, Barton, and Rice Counties, Central Kansas.

<479> CONT.
CONL/SUB-75/30203/1; 80 pp. (1975, June)

The Hutchinson Salt Member of the Permian Wellington Formation is described in a five-county study area of 8,000 square miles. The salt shows a median thickness of 250 feet northwestward from a reference section in a cored test hole near Hutchinson, Kansas. The salt is regionally thinner to the northeast in Lincoln County and regionally thicker to the southwest. Locally the salt is thin on anticlines coinciding with underlying oilfields and thicker in synclinal areas due to depositional variations. Depth to salt varies from 800 feet on the east to 1500 feet in southwestern Barton County. Most of the 22,200 oil and gas test holes in the study area have been drilled with fresh water, causing dissolution of the salt during drilling, commonly resulting in borehole enlargement to three times the diameter of the drill bit. Some older rotary drilled holes have borehole enlargement to 10 feet in diameter. After drilling ceases, no salt dissolution occurs in oil and gas test holes which have properly cemented surface casing protecting all aquifers above the salt. This conclusion is based on a review study of the hydrology of central Kansas aquifers, the examination of cased hole logs some of which show shifts indicative of a fluid-air interface outside the casing, and a study of re-entered boreholes. Post-drilling salt dissolution does occur in holes abandoned with no surface casing left in place (illegal after 1935) as demonstrated by two reentered holes, one with a calculated borehole diameter of 8-1/2 feet in the salt. Extensive dissolution of the Hutchinson Salt in oil and gas test holes in central Kansas is a rare and unusual event; in only seven known instances (six of which are within the study area) did such dissolution lead to collapse and surface subsidence. With an estimated 72,000 holes drilled through the Hutchinson Salt Member within the State of Kansas, this is a ratio of approximately one occurrence for every 10,000 oil and gas test holes. (Auth) (LKM)

Stratigraphic Unit Thickness; Depth to Water Table

SALT DEPOSITS; GEOLOGY; HYDROLOGY; LOGGING, WELL; LOGGING, GAMMA; LOGGING, ELECTRIC; LOGGING, RESISTIVITY; LOGGING, SONIC; MAPS; AQUIFERS; STRATIGRAPHY; CORES; DRILLING; SUBSIDENCE; BOREHOLES; FIELD STUDIES

<480>
Warner, D.L., U.S. Department of the Interior, Cincinnati Water Research Laboratory, Cincinnati, OH.

Subsurface Disposal of Liquid Industrial Wastes by Deep-Well Injection.

American Association of Petroleum Geologists, Memoir 10; Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 11-20), 253 pp. (1968, August)

Deep-well injection is one method of disposing of concentrated, unusable, relatively untreatable liquid wastes. Feasibility of deep-well injection depends on geologic and engineering suitability of sites, the volume and physical and chemical characteristics of the waste, economics of constructing and operating an injection system, and legal considerations. A suitable injection zone must have sufficient permeability, porosity, thickness, and areal extent to act as a liquids storage reservoir at safe pressures, and it must be below and confined vertically from freshwater circulation by impermeable rocks. Other considerations in site selection include geometry of the injection zone, aquifer hydrodynamics, mineralogy and chemistry of reservoir rocks and interstitial fluids, earthquakes and possible fault movement, abandoned and improperly plugged wells penetrating the injection zone, injection pressures and possibilities of hydraulic fracturing, and temperatures and pressures within the reservoir. Generally suitable sites should be found in thick sedimentary, saltwater-bearing, marine sequences in which the geology is relatively well known, such as in the synclinal basins discussed in this memoir, or along the Atlantic and Gulf Coastal Plains. Suitability of waste for deep-well injection depends on the physical and chemical properties of both the waste and the reservoir walls and fluids. Limiting factors within the well and reservoir include volume, injection pressure, intake rate, and the possible interactions between wastes, reservoir rocks, and fluids that could cause plugging of the pore spaces. A review is provided of the 16 states that have a total of 170 waste injection systems, the kinds of wastes injected, the years wells were placed in operation, the range of well depths, and the geologic formations of wells. Careful geologic evaluation is a prerequisite for deep-well waste injection. (Auth) (DCW)

Principles discussed in the paper are specific to liquid industrial wastes other than liquid radioactive wastes, though the principles also apply to liquid radioactive wastes. This paper provides 58 references.

WELLS, INJECTION; SITE SELECTION; WASTE DISPOSAL; WASTE STORAGE; WASTE VOLUME; WASTES, COMMERCIAL; WASTES, LIQUID; WASTES, RADIOACTIVE; WASTE-ROCK INTERACTIONS; MINERALOGY; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; GEOLOGIC FORMATIONS; GEOLOGIC STRUCTURES; BASINS, STRUCTURAL; GEOLOGY; ROCKS; SEDIMENTARY ROCKS; METHODS; AQUIFERS; HYDRAULIC FRACTURING; HYDRODYNAMICS; PRESSURE EFFECTS; PERMEABILITY; POROSITY; EARTHQUAKES; FAULTS; FRACTURES; ECONOMICS; INJECTION

<481>
Waters, K.H., S.P. Palmer, and W.H. Farrell, University of California Department of Materials Science and Mineral Engineering, Berkeley, CA; Systems, Science, and Software, Inc., La Jolla, CA.

Fracture Detection in Crystalline Rock Using Ultrasonic Shear Waves.

LBL-7051; 46 pp. (1978, December)

An ultrasonic shear wave reflection profiling system for use in the detection of water-filled cracks within a crystalline rock mass is being laboratory tested. The first priority is to develop a source-receiver pair that will enable the detection of seismic energy reflected from the back face of a 1.0 m X 0.65 m X 0.30 m rectangular granite slab. Direct observation of the primary reflection and its first multiple has been possible

<081> CONT.

using a source-receiver pair that broadcasts and receives horizontally polarized shear (S sub H) waves. Because shear (S) waves should not propagate in a fluid or into a fluid-filled crack, it was concluded that S sub H wave reflection techniques should theoretically yield the best results in the detection of waterfilled cracks. Experiments testing these techniques were performed on an irregular tensile crack induced approximately 0.5 m below one circular face of a 1.0 m diameter, 1.8 m long granite cylinder. Again good reflection data were obtained with the crack either air filled or water filled. During this last study, data were collected which suggested a frequency dependent horizontal shear wave reflection coefficient for a granite-water interface. Surface waves can severely hinder the detection of reflected events, so two techniques of reducing this "noise" were investigated. The first uses physical obstruction (such as a slit trench) to scatter the surface waves; the second, used in petroleum exploration, uses a linear array of receivers located on the free surface to cancel waves propagating parallel to the array, thus enhancing waves with propagation vectors orthogonal to the linear array (reflected events). Deconvolution processing, which is the convolution of the inverse of the source input with the received output, was found to be another method useful in surface wave cancellation. The results of these experiments are very encouraging, and further research should include a small-scale field test using shear wave reflection techniques. (Auth) (LKM)

SEISM C REFLECTION; METHODS; FRACTURES; GRANITES; IGNEOUS ROCKS; LABORATORY STUDIES; FAULTS; JOINTS

<082>
Beart, W.D., Sandia Laboratories, Albuquerque, NM.

New Mexico Waste Isolation Pilot Plant - A Status Report.

SAND-76-5992; CONF-761020; Waste Management '76, R.G. Post (Ed.), Proceedings of a Symposium, Tucson, AZ October 3-6, 1976, (pp. 173-187), 370 pp. (1976, October)

The major activity during the past year and a half has been selection and evaluation of a specific site for the WIPP. The area required must be composed of 6 sq km for the repository itself and 70 sq km for the surrounding control zones. The site must be located so that a salt dissolution front is not likely to breach the repository within 0.5 million years; structure should be predictable, with only minor deformations and few or no fractures; suitable salt horizons for a repository must be located between 300 and 900 m depth, and be adequately isolated from over- and underlying aquifers; currently, drill holes through the evaporites are not permitted within the area of the repository or a 1.6 km-wide zone around it. After a renewed investigation of the originally selected site in 1975 revealed structural deformation with dips over 70 degrees, fracturing, and accumulations of pressurized brines in the basal Salado Formation, a new site was located 10 km to the SW. Although evaluation is continuing, evidence to date indicates a favorable repository site. Conflicts of the repository with natural resource development are being investigated. Potash exploratory holes are being drilled. Several petroleum exploratory holes are being drilled in the region around the site; the ones closest to the repository are non-producing. A Draft Environmental Impact Statement is being prepared by Sandia. Site evaluation studies, including the resource assessment, will be completed in early 1977. Public hearings on the environmental impact could commence in late 1977 or early 1978. Favorable hearings would allow funding for construction starting in 1979, with initial waste acceptance possible by 1981. Quantities, sources, and character of ERDA waste that the WIPP will be required to accept have not been firmly established. ERDA TRU waste available for potential storage at WIPP, beginning in the mid-1980's, will be 10(E-5) cu ft/yr from Rocky Flats, 5000 cu ft/yr from LASL (plus 50,000 cu ft backlog), 4 X 10(E+5) cu ft/yr from INEL, and about 2 X 10(E+6) cu ft from Hanford and Savannah River. High-level waste will include 2.7 X 10(E+5) cu ft from Savannah River, 1.1 X 10(E+6) cu ft from Hanford, and 6 X 10(E+5) cu ft from INEL. The pilot-scale operation will accomodate only TRU and intermediate-level wastes, with experiments with typical commercial high-level waste. All waste will be solid. ERDA's policy of engineered retrievability is expected to pose a serious problem for the WIPP design concept, particularly with regard to intermediate- and high-level wastes. (AUTH) (LKM)

Waste Volume

WIPP; REPOSITORY; SALT DEPOSITS; DISSOLUTION; SITE SELECTION; GEOLOGIC STRUCTURES; FRACTURES; BRINES; DRILLING; POTASH; HALITE; EVAPORITES; ENVIRONMENTAL IMPACT STATEMENTS; WASTES, TRANSURANIC; WASTES, HIGH-LEVEL; WASTES, SOLID; RETRIEVABILITY; DESIGN; STORAGE, GEOLOGIC; WASTE STORAGE; WASTE DISPOSAL; FIELD STUDIES

<083>
Weaver, C.E., Georgia Institute of Technology, School of Geophysical Sciences, Atlanta, GA.

Fine-Grained Sheet Silicate Rocks.

T/OBI/SHE-7009/8; 79 pp. (1977, September)

In the first part of this report the author reviews the sedimentary petrology of the fine-grained terrigenous clastics, and proposes the term "physil" (an abbreviation of phyllosilicate) to describe all sheet silicate minerals regardless of size. Physils are unconsolidated materials composed predominantly of sheet silicates. Physillites are indurated rocks composed primarily of sheet silicates. The second part of the report recommends physilic rocks suitable for evaluation as high-level radwaste repositories. The water content, porosity, permeability, and mineralogy of fine-grained sediments must be taken into account when considering them for a repository. The effect of quartz, feldspar, carbonate, and organic material must also be considered. The three major groups of physils are smectite, mixed-layer illite-smectite, and illite-chlorite. The interlayer and pore water contents decrease as the proportion of smectite decreases. The physillites can be classified into 10 types; only 8 of these are volumetrically important. All are 20-80%

<483> CONT.
quartz and 10% feldspar plus quartz, the remainder being either illite-chlorite silt, mixed-layer silt or clay, or smectite clay. There are 6 basic varieties of calcareous physilic rocks, of which only 2 occur in abundance: those of approximately 30% carbonate, 20% quartz, and 50% physils (illite-chlorite), and those of approximately 60% carbonate, 7% quartz, and 33% physils (illite-chlorite). The feldspathic physilic rocks can be narrowed down to 2 or 3 basic types: 30-60% physils (illite), quartz/feldspar ratio = 1.0-1.5; 10-30% physils (illite), quartz/feldspar ratio = 1.0-1.5; and (?) 20-50% physils (smectite), quartz/feldspar ratio = 1.0-1.5. The quartzose physilic rocks can be classified into 20 types, only 4 of which are important: 40-50% quartz, 0-10% carbonate, illite-chlorite silt; 40-50% quartz, 0-10% carbonate, mixed-layer silt; 50-90% quartz, 0-10% carbonate, mixed-layer silt; and 40-50% quartz, 0-10% carbonate, smectite clay. Only one organic-rich physilic rock type need be considered (the Chattanooga Shale). Forty formations are listed that contain rock types of the mineralogies recommended for testing. Some of the more prominent of these are the Chemung and Catskill (PA), Atoka and Springer (OK), Pierre (SD, NT, CO, WY), Bolt (WY), Sequatchie and Chickamauga (TN), Miocene sediments (GA), and Triassic sediments of (CT, NC, SC). The 13 most abundant rock types could be collected from 5 to 6 formations. (Auth) (LKH)

MINERALOGY; SEDIMENTARY ROCKS; STORAGE, GEOLOGIC; EVALUATION; SHALES; CLAYS; SILTS; SILTSTONES; ILLITE; CHLORITES; SMECTITE; CYCLOSIC FORMATIONS; REVIEWS; LITHOLOGY; GEOLOGY

<484>
Wenner, D.B., University of Georgia, Athens, GA.

Supplementary Report: Geologic Feasibility of Talc and Serpentinite Bodies from the Appalachian Mountain Region of Eastern United States with Regard to Siting of Radioactive Waste Repositories.

Y/OWI/SUB-4696/2; 27 pp. (1976, January)

This report is a follow-up investigation of certain aspects discussed in an earlier report titled "Geologic Feasibility of Talc and Serpentinite Bodies from the Appalachian Mountain Region of Eastern United States with Regard to Siting of Radioactive Waste Repositories." Further information on the suitability of ultramafic rocks for storage of radioactive wastes was obtained through interviews with Dr. S.L. Morency of the Laboratorie de Geochemie, Department des Sciences de la Terre, the University of Quebec at Montreal, and by field studies at selected localities in Maryland and Vermont. Information provided by Dr. Morency indicates that serpentine is stable to 600-700 C, although equilibrium stability temperatures are lower (about 500 C, using DTA analysis). Serpentinized ultramafic bodies of large dimensions occur in a wide range of locations in the United States and Canada. Ultramafic rocks in general have a relatively low thermal coefficient of expansion, compared to many other rock types (i.e., granite). Serpentine has the ability to undergo self-sealing at depth via flowage and small-scale shearing, thus preventing groundwater influx. Ultramafic bodies that typically range in composition from unserpentinized dunites and peridotites to pure serpentinites offer a wide variation in physical properties and thus offer a variety of geological settings for containment of radioactive wastes. Finally, field studies in Vermont and Maryland verified that very little groundwater inflow occurs in talcose rocks except in large fault zones. Even in these instances, only very slight seepage occurred. A major drawback to the usage of talc for containment of radioactive wastes is the relatively small size of most bodies. However, some of the large talc-carbonate units in the Ammonoosuc Valley Region of Vermont may be useful if adequate subsurface information can be obtained. It is recommended that consideration also be given to other crystalline rock types, particulary schists, which contain micaceous minerals with properties similar to talc and serpentine. Laboratory investigations of rock properties, especially thermal conductivity, thermal expansion coefficient, thermal stability, and the effects of high radiation fluxes, are also recommended. (LKH)

SERPENTINE; TALC; IGNEOUS ROCKS; METAMORPHIC ROCKS; SITE SELECTION; WASTE STORAGE; WASTE DISPOSAL; STORAGE, GEOLOGIC; THERMAL PROPERTIES; THERMAL CONDUCTIVITY; THERMAL EXPANSION; ULTRAMAFICS; SEEPAGE; FAULTS; FRACTURES; SCHISTS; MICA; FIELD STUDIES; REVIEWS

<485>
Wilcox, R.E., R.B. Kaser, R.H. Pilger, Jr., P.C. Ferrell, Jr., R.A. Barlow, and J.K. Whisman, Louisiana State University, Institute for Environmental Studies, Baton Rouge, LA.

Regional Geology.

Y/OWI/SUB-4112/37; An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 219-352), 475 pp. (1977, September 30)

A number of geological, geophysical, and geochemical investigations were undertaken to more fully characterize the structural and stratigraphic characteristics of the North Louisiana Salt Dome Basin. As part of the project over 500 structural and stratigraphic maps of the salt dome basin were constructed. Calculated growth rates for the basin are less than 0.1 mm/yr and average growth rates over long periods of time are less than 0.05 mm/yr. Maximum inferred rates for any dome are: Cenozoic, 0.03 mm/yr; Upper Cretaceous, 0.02 mm/yr; and Lower Cretaceous, 0.04 mm/yr. From additional seismic data the base of the Vacherie dome is connected in all probability to a layer between 16,000 and 17,000 ft below sea level. Other data also suggest the Vacherie dome to be elliptical in shape while the Rayburn's and Prothro are inferred to be more circular. Geochemical work has begun to show the mineralogy and chemical composition of Gulf Coast salt-dome salt. Mineral resources maps were prepared for the Vacherie and Rayburn's salt dome areas. (NBV)

Numerous diagrams and calculations are used to explai models of salt dome growth and regional prole . Tables presenting basic data and computations for selected salt domes showing growth rates are included. Table V-6-1 gives the existing oil and gas wells in the Vacherie dome and Rayburn domes areas. See appendices for more data.

<485> CONT.
CHEMICAL COMPOSITION; COMPUTER PROGRAMS; EQUATIONS; GEOCHEMISTRY; GEOLOGIC STRATA; GEOLOGIC STRUCTURES; GEOLOGY; GEOPHYSICAL SURVEYS; LABORATORY STUDIES; MINERALS; ROCK MECHANICS; GEOLOGIC HISTORY; SEISMIC REFLECTION; SEISMOLOGY; WELLS; SALT DOMES

<486>
Wiley, J.R., Savannah River Laboratory, Aiken, SC.

Ion Exchange Decontamination of Alkaline Solutions of Nuclear Waste.

DP-MS-76-69; CONF-770301; American Chemical Society Meeting, New Orleans, LA, March 20-25, 1977, (17 pp.). (1977)

An organic ion exchange process has been developed at Savannah River Plant to remove Cs 137, Sr 90, and Pu from alkaline salt solutions produced by reprocessing of nuclear fuels and defense waste. Waste tank supernate slurried with sludge is filtered, and the resulting solution passed through three 2.5 liter columns. The first two columns are filled with Duolite ARC-355, a phenolsulfonic cation exchange resin, the third with Chelex 100, an iminodiacetate chelating resin. About 775 liters of waste were decontaminated in 9 tests, with batch sizes ranging from 50-150 liters. Flow rates were 2.5-7.5 liters/hr. Cs 137 decontamination factors were from 8.0 X 10(E+5) to 2.9 X 10(E+5), Pu from 5 to 300, and Sr 90 from 5 to 7700, the latter using various sorbents. Sr 90 decontamination factors using Chelex 100 were generally over 1000. Product activity did not correlate with the type of sludge mixed with the waste solutions, indicating that colloids were not forming and carrying radionuclides through the exchange columns. Estimated residual activites in a salt cake made by evaporating the decontaminated solution are 3.7 nCi/g for Cs 137, 0.6 nCi/g for Sr 90, and less than 1.7 nCi/g for Pu. After decay of Ru 106, residual activity would be about 6 nCi/g. In these tests, the radionuclides remaining after ion exchange were fixed on 100 ml of zeolite, representing an overall volume reduction of 1500 times. (LEM)

Total Ion Concentration

Cs 137; Sr 90; Pu

ION EXCHANGE; RESINS; WASTES, HIGH-LEVEL; VOLUME REDUCTION; WASTES, LIQUID; SALTS; CESIUM 137; STRONTIUM 90; PLUTONIUM; WASTE TREATMENT; POLYMERS; SLURRIES; ZEOLITES; FIXATION; CHELATES; ORGANIC COMPOUNDS

<487>
Witherspoon, P.A., and O. Degerman, Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA; Swedish Nuclear Fuel Supply Company, Stockholm, Sweden.

Swedish-American Cooperative Program on Radioactive Waste Storage in Mined Caverns in Crystalline Rock, Program Summary.

LBL-7009; SAC-01; 26 pp. (1978, May)

In late Spring of 1977, the Swedish-American Cooperative Program was initiated to investigate the geological, geophysical, hydrological, geochemical, and structural effects anticipated from the use of a large crystalline rock mass as a geologic repository for nuclear waste. The report details the cooperative program of investigations at the Stripa mine, an iron mine in a massive body of granite, located about 150 km west of Stockholm. The Swedish Nuclear Fuel Safety Program carried out studies concerning the properties of the Stripa Granite, including rock stress measurements, a study of the pressure and temperature on rock permeability, thermal stress, and water analyses. The Lawrence Berkeley Laboratory (LBL) made a fracture hydrology assessment study in order to define the surface and subsurface hydrological conditions of the fractured granite at Stripa and to determine how these conditions vary as a function of time and temperature. In addition to this study, LBL also conducted two heater experiments to permit the investigators to determine the temperature effects in granite caused by the storage of nuclear waste. (JET)

IGNEOUS ROCKS; GRANITES; DISPOSAL SITE; WASTE STORAGE; PERMEABILITY; GEOLOGY; GEOCHEMISTRY; WASTES, RADIOACTIVE; TEMPERATURE; HYDROLOGY; REPOSITORY; HEAT TRANSFER; SITE EVALUATION; STORAGE, GEOLOGIC; MINES

<488>
Woodward-Clyde Consultants, San Francisco, CA.

Preliminary Geologic Site Selection Factors For the National Waste Terminal Storage Program.

Y/OWI/SUB-76/16531; 59 pp. (1977, June)

To assist in the National Waste Terminal Storage (NWTS) Program efforts in locating, constructing, and operating federal repositories for the safe geologic isolation of radioactive wastes, the geologic considerations and associated factors pertinent to the selection of repository sites are described in detail. Siting considerations involve depth, volume, tectonic stability, and hydrologic regime of the host rock. A description of the information necessary to assess the geologic factors and how this information can be obtained, and an illustration of a general approach of how the geologic factors could be applied and integrated to assess the acceptability of candidate sites are also presented. Charts and tables of geologic siting objectives and criteria are included. (CAJ)

SITE SELECTION; REPOSITORY; STORAGE, GEOLOGIC; WASTE STORAGE; WASTES, RADIOACTIVE; SITE SURVEILLANCE; RECOMMENDATIONS; GEOLOGY; ROCK MECHANICS; HYDROLOGY; THERMAL PROPERTIES; PHYSICAL PROPERTIES

<489>
Yardley, D.H., University of Minnesota, Minneapolis, MN.

Hydrology of Some Deep Mines in Precambrian Rocks.

Y/OWI/SUB-8367/1; 72 pp. (1975, October)

<489> CONT.

A number of deep mines in Precambrian rocks of the Lake Superior region were investigated during the summer of 1975 as part of a preliminary study of the dryness, or lack of dryness, of these rocks at depth. Mines investigated included: Homestake Copper, Centennial Mine, near Hancock, MI; White Pine Copper Mine, Ontonagon Co., MI; Sombra Mine, Tower, MN; Geco Mine, Manitouwadge, Ontario; George MacLeod Mine, Ontario; Denison Mine, Elliot Lake, Ontario; and Creighton Mine, Sudbury, Ontario. Details of mine geology, hydrology, and structural and stratigraphic sections are provided in the form of attached documents provided by the mining companies. Mine pumping records and regular geologic mine maps were found to be inadequate because the amount of water encountered in deep workings is so minute that noting it on plans and sections serves no purpose for the mine operators. No evidence of running, seeping, or moving water was seen or reported at depths below 3000 ft; above 3000 ft, water seepages do occur in some of the mines. Rock movements associated with extensive mining should increase the secondary permeability of the rocks adjoining the mined out zones. Also, most ore bodies are located where there have been more than average amounts of faulting and folding in the geologic past. In general, Precambrian rocks of similar geology to those seen, well away from zones disturbed by deep mining or extensive geologic activity, should be even less permeable. (LKN)

MINES; IGNEOUS ROCKS; METAMORPHIC ROCKS; WATER; GROUND WATER; MOISTURE; SEEPAGE PITS; REVIEWS; FIELD STUDIES; ORE DEPOSITS; FRACTURES; FAULTS; REPOSITORY; SITE SELECTION; STORAGE, GEOLOGIC

<490>

Zeigler, D.L., J.W. White, R.J. Johnson, L.Q. Feaq, R.R. Tetar, and S.Y. Chang, Rockwell International, Atomics International Division, Rocky Flats Plant, Golden, CO.

Status Report, Waste Incineration and Fixation for Waste Management, Production, and Reprocessing Division of the Department of Energy (July-December 1976).

RFP-2655; 20 pp. (1977, November)

An incineration-fixation system is being developed for treatment of general plant waste based on the existing pilot-plant fluidized bed incinerator and vitrification of residues. A new waste clarifier is being evaluated for separation of trap metal (about 0.5 wt%) from the waste. Volume reductions of 4.8 to 1 and weight reductions of 2 to 1 were obtained by incineration of HEPA filters. A tributyl phosphate-solvent mixture was burned in the incinerator using a sodium carbonate-catalyst mixed bed to tie up the phosphate as sodium phosphate; two runs averaged 103.9% phosphate recovery. Vitrification studies were conducted on the simulated incinerator ash and noncombustible waste stream from a LWR nuclear fuel reprocessing plant. Joule-heated and standard resistance-type melt reactors have been designed and are being fabricated. Various castings of vitrified waste were evaluated; the method that appears to be best uses a glob-cutter or plug valve at the melt reactor outlet, with the molten glass dropping onto a moving plate. The product is 18 cm in diameter by 6 cm in height; optimum annealing time is being determined. Computer modeling of He generation from Pu in the waste pellets indicates that the probable maximum Pu oxide content of 0.3 wt% will require 2000 years before undiffused He exceeds its solubility at the glass yield strength. (LKN)

INCINERATION; IMMOBILIZATION; WASTES, SOLID; FILTERS; GLASS; RESINS; VOLUME REDUCTION; EQUIPMENT; METHODS; VITRIFICATION; HELIUM; RADIOLYSIS; WASTE MANAGEMENT; WASTES, LIQUID; WASTES, RADIOACTIVE; PLUTONIUM

<491>

Zeller, E.J., E.E. Angino, and D.P. Saunders, University of Kansas, Department of Geology, Lawrence, KS; Texas Instruments, Inc., Geophoto Division, Dallas, TX.

Antarctica, A Potential Disposal Site for the World's High-Level Radioactive Waste.

Modern Geology 6:31-36. (1976)

The potential of large masses of ice for the disposal of radioactive wastes is discussed. As long as the average temperature of ice is maintained well below the freezing point it provides many of the same features which make salt an attractive storage medium. Fractures are self-healing through recrystallization and plastic flow and it is impermeable to water. The heat conductivity of ice is lower than that of salt but it is comparable to common rock types such as granite. In Antarctica, ice occurs in very large continuous masses of high purity. Surface temperatures are very low and the central portion of Antarctica is the largest area on earth which is essentially devoid of flora or fauna. The interior of the ice cap in East Antarctica is the most isolated region from normal human contact. If hot containers of high-level radioactive waste are placed on top of the ice sheet they can melt their own emplacement shafts through the entire thickness of the ice. Ultimately, the containers would come to rest at the ice-rock interface from one to two kilometers below the surface of the ice. Containers would be designed to sink slowly and vertically. Snow, melt water and plastic flow would permanently seal the shafts behind the sinking containers. It is recognized that certain problems relating to the stability of the ice sheet are concerned directly with the heat balance at the ice/rock interface. An increase in the thermal input to the interface zone could result in a rapid increase in the rate of movement of the ice sheet over its bed. The mechanisms which might trigger such a movement can be eliminated if the waste canisters are tethered so that they are free only to sink a predetermined distance into the ice. (RAF)

WASTES, HIGH-LEVEL; ICE; CONTAINERS; THERMAL PROPERTIES; REVIEWS; WASTE STORAGE, REPOSITORY; PRECIPITATION, METEOROLOGICAL; ROCKS; SEALING; SITE SELECTION; WASTE DISPOSAL; GLACIERS; STORAGE, GEOLOGIC; DESIGN; STORAGE, ABOVEGROUND; HEAT SOURCES; THEORETICAL STUDIES; SITE SELECTION

<492>

Zeller, E.J., D.P. Saunders, and E.E. Angino,

<492> CONT.
Putting Radioactive Waste on Ice - A Proposal for an International Radionuclide Depository in Antarctica.

Bulletin of the Atomic Scientists 29:8-9, 50-52. (1973, January)

Permanent disposal of high-level radioactive wastes under the Antarctic ice cap is proposed as a solution to the waste management problem. All available evidence indicates that if the wastes were fused into a solid glass matrix, they could be practicably and safely transported to the interior of the ice cap. If placed on the surface of the ice, they would melt their own emplacement shafts which would rapidly reseal by freezing and plastic flow. The containers would sink slowly to the ice-rock interface where they would remain safely isolated from the biosphere for more than 250,000 years. Tentatively selected disposal sites are 600-1000 km inland from the nearest natural flora or fauna, isolated from the sea by mountain ranges as well as by long horizontal thicknesses of ice and frozen ground. The sites were selected to effectively eliminate any harmful effects due to ice movements. The major problems to be solved to establish the feasibility of this approach include complete literature and laboratory studies of the basic principles involved, followed by field surveys and experiments to achieve the following: (1) determine ice-canister interactions; (2) survey detailed ice thicknesses, movements, and sub-ice topography; (3) determine temperature gradients down to the rock-ice interface; and (4) design, construct, and field test both engineering models and prototypes of an operating fail-safe transport and emplacement system. A brief comparison of the ice cap approach with other methods which have been suggested for ultimate disposal indicates that it holds promise of being the most practical and safest solution. (Auth) (LKE)

SITE SELECTION; ICE; GLACIERS; REPOSITORY; WASTE DISPOSAL; WASTE STORAGE; STORAGE, GEOLOGIC; DESIGN; THEORETICAL STUDIES; REVIEWS

<493>
Abrams, L.A., R.L. Schlegel, and R.P. Sullivan, NUS Corporation, Rockville, MD.

As Low as is Reasonably Achievable (ALARA) Studies Relative to the National Waste Terminal Storage Program.

TYDOE/SUB-77/02509; NUS-3028; 96 pp. (1977, October)

The report presents the history of development of the ALARA concept and criteria as applied to both off-site (environmental) and on-site (occupational) exposures at nuclear power plants as well as the current status of activities within the various federal agencies directed toward developing ALARA criteria for other areas of the nuclear fuel cycle. Environmental ALARA criteria related to minimizing radiation to the surrounding population, a set of non-generic, and probably non-numerical criteria, are discussed. The criteria will apply to the various time phases of repository operation and will provide qualitative guidance on procedures, equipment to be used, operational philosophy, monitoring requirements, etc., related to repository operation. A number of studies are being performed by and for the NRC and the EPA which will directly impact on the ALARA criteria being developed, and provide the methodology and technical guidance for their future development. These are generally summarized in tables. A DOE study evaluates nuclear power plant experience and is to develop technical guidelines for a specific power plant. One NRC sponsored study is developing data on dose commitments for all segments of the fuel cycle and another has recommended strengthening of the regulatory base. An interagency study is reviewing the Federal Radiation Council (FRC) guideline limits, and AIP is sponsoring a study of methodology that can be utilized for specific jobs, primarily at power plants. Recommendations are discussed to assure that evolving ALARA concepts are periodically brought up to date and that such concepts be made available to those subcontractors who have responsibility for design and operation of a repository. The rationale for including training programs and audits are given. One of the main objectives of this study is to identify reports, articles, testimony, and other sources discussing ALARA criteria that may be pertinent to the NWTS program. There is an annotated bibliography of some 83 sources from 1970 to the present giving information on ALARA criteria and its application. (CAJ)

REPOSITORY; RECOMMENDATIONS; LICENSING; WASTE STORAGE; WASTES, RADIOACTIVE; WASTE MANAGEMENT; COST BENEFIT ANALYSIS; REVIEWS; EXPOSURE, POPULATION; STANDARDS, FEDERAL; EXPOSURE, OCCUPATIONAL; MONITORING; HAZARD ANALYSIS

<494>
Salestri, E.J., R.W. Torres, S.D. Pahva, and J.P. Brannen, The SDM Corporation, Albuquerque, NM; INTERA Environmental Consultants, Incorporated; Houston, TX; Sandia Laboratories, Albuquerque, NM.

Preliminary Safety Assessment of the Waste Isolation Pilot Plant Facility.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 513-520), 563 pp. (1979)

This paper summarizes the efforts to perform a safety assessment of the WIPP facility being proposed for southeastern New Mexico. Consequences of the events which result in loss of containment are of fundamental importance when making site selection, design decisions, and waste form acceptance criteria. This assessment of the WIPP facility is limited to consequences in terms of the dose to a maximally exposed individual as a result of introducing the radionuclides into the biosphere by means of five scenarios, four of which represent a liquid breach of the repository with the subsequent transfer of the waste material through the major aquifer of the region, and one of which represents a breach by resource exploration drilling activity which brings waste directly to the surface. The scenario description includes identification of a breaching event, a physical mechanism for transporting the waste through the breach, and the response of the storage medium to the event. A mathematical model is then constructed describing the transport process in the modified media. This model is utilized to determine the concentration of the various radionuclides as a function of temporal and spatial coordinates in the modeled region. Once the wastes are in the biosphere, an individual's exposure is assumed to come totally from the contaminated material or area. The consequence parameter is the 50-year dose commitment from one year of exposure. Results of the two worst case liquid breach scenarios would give a dose range (in rems) from 2.0 X 10(E-9) to the liver, thyroid, and lung, to 4.5 X 10(E-6) to bone for scenario 1 (a connection between the overlying and underlying aquifers through a failed well shaft), and from 2.4 X 10(E-6) to the lung to 4.9 X 10(E-3) to bone for scenario 4 (entire overlying aquifer flows through the repository). Compared with doses from general sources such as television (1 X 10(E-4) rems) or background at the WIPP site (100 X 10(E-3) rems), these higher (1.5 X 10(E-5) to 2.8 X 10(E-2) rems). The purpose of this analysis was to provide data upon which the safety of the WIPP facility may be decided for contact-handled TRU. No conclusions regarding the safety are made. The extremely low doses received associated with the liquid breaches should be considered in the waste acceptance criteria and canister lifetime requirements, and the high doses received due to drilling could be reduced significantly by choosing an emplacement scheme that prevents several containers from being penetrated by the same borehole. (Auth) (DCW)

WASTES, RADIOACTIVE; WASTES, TRANSURANIC; WASTE STORAGE; WASTE DISPOSAL; TRANSURANICS; RADIATION, BACKGROUND; MODELS; MODELS, MATHEMATICAL; REPOSITORY; WIPP; RADIATION DOSE; SAFETY; AQUIFERS; ENVIRONMENTAL EXPOSURE PATHWAY

<495>
Bischof, W., H. Pelzer, and D. Rasschnig, Goettingen University, Bundesministerium fuer Forschung und Technologie, Bonn, German Federal Republic.

System Study, Radioactive Wastes in the Federal Republic of Germany. Vol 4. The Law Concerning the Disposal of Radioactive Wastes.

<495>

REGULATORY AND ECONOMIC ASPECTS

<495> CONT.
SRA-8; 68 pp.(1977, January)

Presented are the legal aspects of the authority for regulation of the nuclear industry in Germany. The study is divided into two parts: the first deals with legal basis of nuclear law and the second part concerns the enforcement of such laws. Discussed under part one are: 1) problems concerning terminology; 2) radioactive waste disposal; 3) other ordinances which pertain to radwaste disposal; 4) administrative responsibilities; 5) questions concerning liability; 6) legal problems with radwaste disposal in the sea, sea bottom, Antarctica, and space; and 7) regulations in internal law. Recommendations are made for improvement of the existing laws. (NDV)

LEGISLATION; LICENSING; SEA DISPOSAL; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, RADIOACTIVE; SPACE DISPOSAL; STANDARDS, INTERNATIONAL

<496>
Blakeman, E.D., J.D. Jenkins, and E.J. Allen, Oak Ridge National Laboratory, Oak Ridge, TN.

An Overview of Material Control and Accountability Considerations for a Waste Isolation Repository.

CONF-780622; American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (14 pp.).(1978)

The material control and accountability needs of a waste isolation repository are examined. Three levels of control are discussed: (1) item identification and control, (2) tamper indications, and (3) quantitative material assay. A summary of waste characterization is presented, and based on these, plus a consideration of the accessibility of the various types of waste, material control by item identification and accountability (where the individual waste container is the basic unit) is recommended. Tamper indicating procedures are also recommended for the intermediate and low level waste categories. (Auth)

See also ORNL/TM-6162

REPOSITORY; WASTE STORAGE; WASTES, HIGH-LEVEL; CLADDING; WASTES, INTERMEDIATE-LEVEL; WASTES, HIGH-LEVEL; CONTAINERS; DIVERSION; NUCLEAR MATERIALS; TRANSPORTATION

<497>
Kroest, H.C., and J. Pfaffelhuber, Bundesministerium des Innern, Abteilung Sicherheit Kerntechnischer Anlagen, Bonn, German Federal Republic; Bundesministerium des Innern, Referat. Entsorgung und Versorgung von Kerntechnischen Anlagen, Bonn, German Federal Republic.

Function and Licensing Status of the Asse Test Storage Facility.

Atomwirtschaft Atomtechnik 22(12):681-682.(1977, December)

Since its acquisition in 1965, the former Asse II salt mine has been converted by the Gesellschaft fur Strahlenforschung mbH (GSF) into an experimental installation for development, testing and optimization of methods for the deposition and final storage of radioactive wastes. Because the mine was not originally intended for this purpose, the type and quantity of wastes have been carefully defined. However, with the exception of the retrievable wastes, the deposited high-level wastes should remain in the Asse from the beginning until the successful conclusion of the experiments. After the amendation of the Atomic Laws and the Radiation Protection Ordinance, a license would be required according to Sec. 7 Para. 1 of the Radiation Protection Ordinance for the performance of research and development programs in the Asse, however, no statement of intent would be needed under Sec. 9b of the Atomic Laws. (ISH)

STORAGE, GEOLOGIC; SALT DEPOSITS; MINES; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; LEGISLATION; LICENSING; REPOSITORY; SAFETY

<498>
Cohen, J.J., R.L. Braun, L.L. Schwartz, and H.A. Toews, Lawrence Livermore Laboratory, Livermore, CA.

Economic and Environmental Evaluation of Nuclear Waste Disposal by Underground In Situ Melting.

Transactions of the American Nuclear Society 18:194-195; UCRL-51713; CONF-780608; American Nuclear Society Meeting, Philadelphia, PA, June 23-28, 1978, (14 pp.).(1978, November 30)

Disposal of high-level nuclear waste by the Deep Underground Melt Process (DUMP) was reviewed and evaluated relative to other proposed methods for waste management. Advantages listed were: 1) elimination of reduction of many processing operations and their attendant risks, 2) potential for safe disposal of all levels of liquid waste, including intermediate and low level, 3) elimination of radioactive waste transportation and its attendant risks, 4) permanent binding of the radioactive waste in a rock matrix deep underground, and 5) assurance of permanent safety from floods, sabotage, or warfare. Problem areas discussed were possible policy conflicts, misconceptions, and technical problems. Although the potential environmental impact (population exposure to radiation) for the DUMP concept was found to be lower than that for alternative disposal methods, the difference was considered of small significance. However, a potential 80% cost savings relative to other geologic disposal methods was indicated. (Auth)(JTE)

WASTES, HIGH-LEVEL; COST BENEFIT ANALYSIS; ECONOMICS; HAZARD ANALYSIS; IMMOBILIZATION; WASTE-ROCK INTERACTIONS; SAFETY; SABOTAGE; FLOODS; EVALUATION; STORAGE, GEOLOGIC; WASTE DISPOSAL; WASTES, LIQUID; EXPOSURE, POPULATION; RADIOACTIVE DECAY; SOLIDIFICATION; HEAT SOURCES; ROCKS; CAVITIES; REVIEWS

<499>
Control Data Corporation.

National Waste Terminal Storage Program, Configuration Management Plan, Volume 1. Management Summary.

<499> COST.
Y/OWI/SUB-77/16535/3; 7 pp. (1977, May)

The Office of Waste Isolation Configurations Management System is a standardized management discipline based on industry standard methodology and the Union Carbide Corporation, Nuclear Division (UCC-ND) procedure designed for use in uranium capacity expansion projects. The plan is to provide the management discipline through which the program cost and schedule trade-off decisions made concerning the site selections and facilities performance and support are communicated and controlled by the OWI. It will be made up of four functions: 1) Configuration Identification, 2) Configuration Control, 3) Configuration Status Accounting, and 4) Configuration Audit. (CAJ)

COST BENEFIT ANALYSIS; RECOMMENDATIONS; QUALITY ASSURANCE; STANDARDS, FEDERAL; WASTE STORAGE; SITE SELECTION; URANIUM

<500>
Davis, W., Jr., R.E. Blanco, B.C. Finney, G.S. Hill, R.E. Moore, and J.P. Witherspoon, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN; Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Correlation of Environmental Impact with Cost of Waste Treatment.

ORNL/NUREG/TM-4; Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing Uranium 233 and Thorium, Section 8.0, (pg. 83-86), 182 pp. (1976, May)

The relationships between the annual costs of the radwaste treatment systems described in previous sections and the impact of radioactive releases, i.e., the dose commitments, from these systems are presented. The annual costs and dose commitments for a base case and succeeding case studies at mid western and coastal sites are summarized in tables. Each cost is the estimated total annual cost required for the additional radwaste treatment systems for a given case beyond that required for the baseline case. Costs are also given as incremental dollars per kilogram of heavy metal for the additional radwaste treatments. Dose commitments are reported on several bases: 1) population annual total body, gastrointestinal tract, bone and thyroid doses out to 55 miles from the plant, which are the total doses from all radioactive materials released from the plant in a specified case; 2) incremental decreases in these annual average values with respect to the baseline case; and 3) maximum annual total-body doses received by an adult at a distance of 1.5 miles from the stack. Factors to convert doses at 1.5 miles to doses at 0.5 mile and 0.1 mile are included. Also given are individual maximum annual and population annual doses to various body organs. Radwaste treatment costs are compared with fuel reprocessing costs, power costs, and total capital investment. (RAF)

COSTS; WASTE TREATMENT; DOSE COMMITMENTS; REACTORS, HIGH-TEMPERATURE GAS-COOLED; REPROCESSING; BODY BURDEN; GASTROINTESTINAL TRACT; BONES; THYROID; RADIATION DOSE; COST BENEFIT ANALYSIS; EXPOSURE, INTERNAL; INHALATION; INGESTION; TRITIUM; CARBON 14; IODINE 129; CESIUM 134; CESIUM 137; KRYPTON 85; POPULATIONS; GASES; STRONTIUM 90; ADULTS; CHILDREN; THEORETICAL STUDIES; MAN

<501>
Davis, W., Jr., R.E. Blanco, B.C. Finney, G.S. Hill, R.E. Moore, and J.P. Witherspoon, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN; Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Correlation of Radioactive Waste Treatments Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing Uranium 233 and Thorium.

ORNL/NUREG/TM-4; 182 pp. (1976, May)

A cost/benefit study was made to determine the cost and effectiveness of various radioactive waste (radwaste) treatment systems for decreasing the release of radioactive materials from a model high-temperature gas-cooled reactor (HTGR) fuel reprocessing plant and to determine the radiological impact (dose commitment) of the released materials on the environment. The study is designed to assist the U.S. Nuclear Regulatory Commission in defining the term "as low as reasonably achievable" as it applies to this nuclear facility. The base case is representative of conceptual, developing technology of head-end graphite-burning operations and of extensions of solvent-extraction technology of current designs for light-water-reactor (LWR) fuel reprocessing plants. The model plant has an annual capacity of 850 metric tons of heavy metal (MTHM, where heavy metal is uranium plus thorium), as charged to about fifty 1000-MW(e) HTGRs. Additional radwaste treatment systems are added to the base-case plant in a series of case studies to decrease the amounts of radioactive materials released and to reduce the radiological dose commitment to the population in the surrounding area. The capital and annual costs for the added waste treatment operations and the corresponding reductions in dose commitments are calculated for each case. In the final analysis, the cost/benefit of each case, calculated as additional cost of radwaste system divided by the reduction in dose commitment, is tabulated or the dose commitment is plotted with cost as the variable. The status of each of the radwaste treatment methods used in the case studies is discussed. Much of the technology used in the advanced cases is in an early stage of development and is not suitable for immediate use in detailed designs of an HTGR fuel reprocessing plant. However, the radwaste treatment methods are believed to be reducible to practice. Four sections of this report have been abstracted separately for the data base. (Auth) (RAF)

COST BENEFIT ANALYSIS; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; REACTORS, HIGH-TEMPERATURE GAS-COOLED; REPROCESSING; DOSE COMMITMENTS; REACTORS, LIGHT WATER; EXPOSURE, POPULATION; FUEL CYCLES; EFFLUENTS; URANIUM 233; THORIUM; CARBON 14; TRITIUM; IODINE 129; IODINE 131; CESIUM 134; CESIUM 137; KRYPTON 85; RADIATION DOSE; THYROID; BONES; STRONTIUM 90; GASTROINTESTINAL TRACT; RUTHENIUM 106; LUNGS;

<501>

REGULATORY AND ECONOMIC ASPECTS

<501> CONT.
RADON GASES; WASTES, LIQUID; EXPOSURE, INTERNAL; EXPOSURE, EXTERNAL; INHALATION; INGESTION; THEORETICAL STUDIES

<502>
Davis, W., Jr., R.E. Blanco, B.C. Finney, G.S. Hill, R.E. Moore, and J.P. Witherspoon, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN; Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Costs for Radwaste Treatment.

CONF/NUREG/TM-8; Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing Uranium 233 and Thorium, Section 6.0, (pp. 68-70), 102 pp.(1976, May)

Costs for the radwaste treatment cases for the 450-metric ton/year model fuel reprocessing plant are estimated as additions to the base case model plant. The capital costs, annual fixed charges, annual operating cost, total annual cost, reprocessing cost, and contribution to the cost of power for the radwaste treatment cases are itemized. Also given are the incremental costs and the corresponding calculated amounts of radioactive materials released (source terms). Annual fixed charges are estimated at 26% of total capital investment; this is typical of cost estimates for investor-owned reprocessing plants. The basis for calculation of the fixed charge rate and the operating cost is detailed. An annual operating expense is added to the annual fixed charge on capital to give the total annual cost of a radwaste treatment case. This cost is then divided by the annual amount of fuel reprocessed, or by the annual amount of electricity that was produced by the reprocessed fuel, to obtain the cost of radwaste treatment per unit weight of fuel reprocessed or the total contribution to the cost of power for each radwaste case. A fuel reprocessing plant with a nominal production rate of 450 metric tons per year heavy metal (U + Th) can service approximately fifty 1000-MW (electrical) high-temperature gas-cooled reactors. (RAM)

COSTS; WASTE TREATMENT; REPROCESSING; REACTORS, HIGH-TEMPERATURE GAS-COOLED; THEORETICAL STUDIES

<503>
Duguid, J.O., and R. Schneider, Battelle Memorial Institute, Office of Nuclear Waste Isolation, Columbus, OH; U.S. Geological Survey, Reston, VA.

Earth Science Technical Plan for Mined Geologic Disposal of Radioactive Waste.

CONF-790602; American Nuclear Society Annual Meeting, Atlanta, GA, June 5, 1979, (8 pp.). (1979, June 5)

The Earth Science Technical Plan (ESTP) classifies the various earth science research tasks involved in developing geologic repositories for radioactive wastes, performed by or on contract for the Department of Energy (DOE) and the U.S. Geological Survey (USGS); shows how these tasks address the principal remaining technical problems related to such repositories; and identifies technical questions that require additional attention. The ultimate objective of the plan is to recommend future program priorities. The ESTP was prepared by DOE and the USGS through a Working Group composed of representatives from the USGS, DOE contractor organizations, and the following components of DOE's National Waste Terminal Storage (NWTS) Program: the Waste Isolation Pilot Plant, the Basalt Waste Isolation Program, the waste isolation program at the Nevada Test Site, and the Office of Nuclear Waste Isolation, Battelle Memorial Institute. In general, the plan assesses the essential program for evaluating the NWTS Program, modifying the current program, improving coordination, and planning future research. Specifically, the plan presents a systematic review of earth science questions concerning geologic repositories and recommends research to answer those questions related to site selection, characterization, and evaluation, repository design and operation, and risk assessment; inventories and classifies current research and development activities to facilitate integration and coordination of DOE- and USGS-sponsored work; identifies research and development activities which should be more closely coordinated for timely resolution of technical questions; specifies technical plans to resolve significant uncertainties regarding mined repositories, such as those discussed in the Interagency Review Group reports and others; provides information that can be used to estimate the time required for resolution of earth science technical questions and to organize the work and resources to determine a realistic schedule for developing a repository; and recommends the preparation of summary documents on key technical problems that will aid in planning future research and development activities. (Auth) (LEM)

GEOLOGY; STORAGE, GEOLOGIC; SITE SELECTION; DESIGN; HAZARD ANALYSIS; WASTE STORAGE; RECOMMENDATIONS; WASTE MANAGEMENT; WASTES, RADIOACTIVE; PROGRAMS, GOVERNMENT; SITE EVALUATION; REPOSITORY; DESIGN; REVIEWS

<504>
Heath, C.A., U.S. Department of Energy, Washington, DC.

State/Federal Licensing Interactions for Radioactive Waste Repositories.

CONF-780622; American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (p. 571); Transactions of the American Nuclear Society 28:571.(1978, June)

The National Terminal Waste Storage Program being conducted by the U.S. Department of Energy (DOE) includes both development of additional technical data concerning ultimate disposal of radioactive wastes and identification of potentially suitable sites at which repositories might be built. The program to identify potentially suitable sites is obviously focusing on the geologic characteristics of various formations in the United States but attention must also be placed on environmental and socioeconomic considerations as well. In the latter two areas especially, there is strong interest on

<504> CONT.
the part of state and local government officials to participate at all stages in the decision-making process leading to the eventual development of a radioactive waste repository at a specific site. Because the development of this joint decision-making process had not been adequately addressed, various events including passage of state laws prohibiting waste disposal have developed. The efforts of DOE to arrive at a satisfactory process for state participation in the program is described. (Complete Text) (Auth)

SITE SELECTION; STORAGE, GEOLOGIC; ENVIRONMENT; ECONOMICS; LEGISLATION; WASTES, RADIOACTIVE; DISPOSAL SITE; LICENSING; REPOSITORY

<505>
Beckman, R.A., S.A. Johnson, and T. Holdsworth, Lawrence Livermore Laboratory, Livermore, CA.

Systems Analysis Study for Waste Management Criteria.

CONF-780850; Waste Management and Fuel Cycles '78, R.G. Post and R.E. Wacks (Eds.), Proceedings of a Symposium, Tucson, AZ, March 6-8, 1978. Arizona Board of Regents, (pp. 139-150), 661 pp.(1978, March 1)

The Lawrence Livermore Laboratory is providing technical support to the U.S. Nuclear Regulatory Commission (NRC) in the development of standards for the management and disposal of high-level and transuranic wastes. The problem logically splits into two phases: the pre-emplacement phase of the waste management system, and the repository post-sealing phase. Using a system analysis approach, the modeling effort was structured in such a way as to produce societal risk evaluations at stated confidence levels so that the NRC can develop regulations for the broadest set of conditions possible. A multicycle approach is used in developing societal risk evaluations. The modeling effort uses a three level concept. At the first level, simple models are developed from first principles of chemistry and physics. These initial models use lumped parameters to provide insight into important processes. The second level modeling effort is designed to provide a flexible, fast running system analysis model. The third level of modeling provides a method for validating the second level models, inputing numerical data, and development of algorithms for use in the second level models. A preliminary assessment of pre-emplacement risks indicates that the risk from the transportation step is the dominating factor. Of lesser importance is the risk from massive loss-of-coolant accidents in the interim storage phase at the fuel reprocessing plant due to a large earthquake. Sensitivity analysis of a deep geologic repository rank the following factors that affect the suitability of such a repository site: 1) radionuclide absorption; 2) permeability; 3) porosity; 4) aquifer length; 5) artesian head; 6) thickness of depository layer; and 7) maximum area of the fracture zone surrounding the access shaft. (Auth) (RAF)

Five figures include illustrations of waste management system analysis model and detailed event tree for truck transportation accidents.

MODELS; WASTE DISPOSAL; WASTE MANAGEMENT; STANDARDS, FEDERAL; WASTES, HIGH-LEVEL; WASTES, TRANSURANIC; HAZARD ANALYSIS; EVALUATION; PHYSICAL PROPERTIES; PREDICTIONS; REPOSITORY; TRANSPORTATION; ACCIDENTS; RADIONUCLIDES; ABSORPTION; PERMEABILITY; POROSITY; AQUIFERS; BASINS, ARTESIAN; FRACTURES; STORAGE, GEOLOGIC; THEORETICAL STUDIES

<506>
Hodge, C.V., and J.E. Campbell, U.S. Nuclear Regulatory Commission, Washington, DC; Sandia Laboratories, Albuquerque, NM.

Calculations of Radiological Consequences from Sabotage of Shipping Casks for Spent Fuel and High-Level Waste.

NUREG-0194; 16 pp.(1977, February)

Radiological consequences of a hypothetical sabotage event which causes a release of radioactive material from a spent fuel cask and a high-level waste cask were calculated, using a modified version of the CORO (Consequence Model) code. Expected values were calculated from a sample of weathering conditions observed over a year's time near the sites of several nuclear power plants. A total of 91 trials were used to obtain the expected value, each trial representing the weather conditions observed at intervals of 4 days and 13 hours. A massive rupture of either a spent fuel cask or a high-level waste cask is considered extremely unlikely in any sabotage event, but a small bore penetration is possible. All gases in the spent fuel, such as krypton are assumed to escape. A small fraction of the solids is assumed to be pulverized into sufficiently fine grained powder to escape as an aerosol. This solid release fraction is treated as a parameter and calculations are reported for values ranging from 10(E-5) to 10(E-2). A solid release fraction of 10(E-2) for a sabotage attack on either a spent fuel cask or high-level waste cask is construed to be uncommonly large. A highly effective sabotage event characterized by a solids release fraction of as much as 10(E-2) is indicated to result in 0 to 2 early deaths and in 40 to 260 latent cancer fatalities. (JMT)

SABOTAGE; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; CASKS; FUEL ELEMENTS; CONTAMINATION; RADIATION HAZARDS; ACCIDENTS; GASES; MODELS; FISSION PRODUCTS; HAZARD ANALYSIS; IMPACT SHOCK; PREDICTIONS; SAFETY; SURFACE CONTAMINATION; TRANSPORTATION; WEATHERING; SPENT FUELS; AEROSOLS; EXPOSURE, POPULATION; THEORETICAL STUDIES

<507>
Hudson, C.R., Oak Ridge National Laboratory, Oak Ridge, TN.

Nuclear Shipping and Waste Disposal Cost Estimates.

ORNL/TM-5976; 22 pp.(1977, November)

Cost estimates for shipping of irradiated fuel, shipping of reprocessing plant waste, and disposal of reprocessing plant waste are determined for five reactor and reactor fuel types - light-water reactor (LWR), mixed-oxide light-water cooled reactor (MOX),

<507> CONT.

Canadian deuterium-uranium reactor (CANDU), fast breeder reactor (FBR), and high-temperature gas-cooled reactor (HTGR). Details of the factors and assumptions used to develop the estimates are provided. A general summary of costs in terms of 1976 dollars/kg heavy metal rounded to the nearest dollar is as follows: irradiated fuel shipping - LWR = $16, MOX = $17, CANDU = $18, FBR = $80, HTGR = $87; reprocessing plant waste shipping - LWR = $8, MOX = $12, CANDU = $3, FBR = $46, HTGR = $33; reprocessing plant waste disposal - LWR = $82, MOX = $85, CANDU = $12, FBR = $109, HTGR = $63. (LKH)

ECONOMICS; TRANSPORTATION; WASTE DISPOSAL; FUEL REPROCESSING; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, TRANSURANIC

<508>

Jet Propulsion Laboratory, California Institute of Technology, Pasadena, CA.

Nuclear Regulatory Commission Program for High Level Waste Management.

JPL Publication 77-59; An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management, Vol. 2, English, T.D. (Project Manager), Appendix D, (pp. D-1 - E-54), 235 pp.(1977, August 12)

The Nuclear Regulatory Commission (NRC) has responsibility under the Atomic Energy Act for regulating management of nuclear wastes at licensed facilities and at disposal sites; under the Energy Reorganization Act, NRC was given additional licensing responsibility for long-term storage and disposal of high-level wastes arising from both ERDA and commercial operations. The National Environmental Policy Act requires the NRC to assess the environmental impacts of proposed waste management regulations. The Commission is working closely with other agencies to assure the soundness of the program for dealing with nuclear wastes. In FY 1976, staffing was begun and work initiated on outlining a program for the regulation of nuclear wastes. The current program consists of the following areas: 1) General Program Planning and Development; 2) High Level Waste Standards Development; 3) HLW Methodology and Data Base Development; 4) HLW Licensing; and 5) Agency Interfaces. Program efforts in FY 1977 will emphasize planning and preliminary program development. In FY 1978, development of the regulatory base will be begun, including standards, criteria, and licensing methodology. NRC has a number of on-going high-level waste management contracts that support the program areas above. For FY 1977 these include: Technical Support in the Development of Nuclear Waste Management Criteria, Assessing Industry Awareness of Issues Surrounding Nuclear Wastes, Parametric Study of Management of Wastes from LWR Fuel Reprocessing Plants, and Risk Assessment Methodology Development for High-Level Waste Isolation. In the area of repository site selection, ERDA is planning to select two high-level waste repository sites by 1978; NRC plans to propose repository suitability criteria by January 1978 and finalize the criteria by September 1978. A major factor in successfully meeting this goal will be the duration and outcome of public hearings on the proposed sites. (Auth) (LKH)

LICENSING; WASTE MANAGEMENT; WASTE DISPOSAL; WASTE STORAGE; DISPOSAL SITE; WASTES, HIGH-LEVEL; STANDARDS, FEDERAL; SITE SELECTION; PLANTS, WASTE TREATMENT; WASTE TREATMENT

<509>

Hirose, K., University of Tokyo, Tokyo, Japan.

Radioactive Waste management in Japan.

CONF-751004; Nuclear Safety, 1975, D.L. Hetrick (Ed.), Proceedings of an American Nuclear Society Topical Meeting, Tucson, AZ, October 5, 1975, (pp. 315-329).(1975)

The present status of radioactive waste management in Japan is reviewed, and some research and development programs are described. The Japanese A.E.C. considers that the treatment of radioactive wastes should be done by operators of nuclear facilities on-site. It is hoped to extend the temporary guide (1973) for cement-solidification of low-level wastes for ocean dumping tests to apply to multi-stage designs. Research and development programs on waste management, especially for medium- and high-level wastes, are being conducted with close cooperation between governmental and industrial organizations. Environmental and social impacts of off-site management of high-level and alpha-active wastes will be very important. The problems of what kind of responsible organization will be most appropriate and how the costs of management should be allocated will become controversial. (CAJ)

WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, RADIOACTIVE; RECOMMENDATIONS; ALPHA PARTICLES; ENVIRONMENT; REACTORS; REVIEWS; CEMENTS; SOLIDIFICATION; ECONOMICS

<510>

La Porte, T.R., University of California, Institute of Governmental Studies, Berkeley, CA.

Nuclear Waste: Increasing Scale and Sociopolitical Impacts.

Science 201:22-28.(1978, July 7)

The article argues that insufficient attention has been paid to the operational aspects of the U.S. radioactive waste management system when it grows to the scale necessary to handle wastes produced by a fully deployed plutonium economy. Without such information, many of the unsettling speculations which have become part of the public debate and are summarized herein cannot be clearly addressed. The article then outlines the types of information necessary to begin estimating the costs and consequences of radioactive waste management. Finally, an index of social exposure to radioactive hazard is proposed to improve the basis for policy decisions in this area. (Auth)

ECONOMICS; FUEL REPROCESSING; INFORMATION; LEGISLATION; PERSONNEL; PLUTONIUM; WASTE MANAGEMENT; WASTES, RADIOACTIVE

<511>
Lash, T.R., U.S. Congress, House of Representatives, Committee on Interior and Insular Affairs.

Statement of Dr. Terry R. Lash.

Oversight Hearings before the Subcommittee on Energy and the Environment, 95th Congress, First Session, Washington, D.C., May 16, 17, and 20, 1977, Serial No. 95-15, (pp. 262-287), 315 pp. (1977)

The principal conclusion of a Natural Resources Defense Council study is that the federal government has not adequately managed existing radioactive wastes and that it is not adequately planning for the ultimate disposal of either military or commercial wastes. In the presentation the hazards of nuclear wastes are briefly discussed. One of the chief concerns is that permanent storage of wastes in retrievable surface storage facilities will be permitted due to lack of proper planning. A number of reports and investigators are cited expressing such concerns. Due to the human factor it may not be possible to have an acceptable safe waste disposal scheme identified and implemented. Technologies are not self-implementing. (NDW)

CONTAINMENT; ENVIRONMENT; LEGISLATION; REGULATIONS; REVIEWS; STANDARDS, FEDERAL; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; TAILINGS

<512>
Lewis, H.W., R.J. Budnitz, W.D. Rowe, H.J.C. Kouts, F. Von Hippel, W.B. Loewenstein, and F. Zachariasen.

Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission.

NUREG/CR-0400; 66 pp. (1978, September)

The Risk Assessment Review Group was organized by the U.S. Nuclear Regulatory Commission in July 1977 to provide advice and information to the Commission on the final report of the Reactor Safety Study, WASH-1400, the "Rasmussen Report". While praising the study's general methodology and recognizing its contribution to assessing the risks of nuclear power, the Review group was critical of the Executive Summary, the procedure followed in producing the final report and the calculations in the body of the report. The Review Group found WASH-1400 to be incomprehensible and difficult to follow, thus making a peer review very difficult. The Executive Summary does not describe the contents of the report and has lent itself to misuse of reactor risk data. The group found areas of conservation and nonconservation in the probability calculations in WASH-1400, with greater uncertainty in the overall probability of a core melt. The fault-tree/event-tree methodology was found to be sound and further development and use was recommended. (BAP)

HAZARD ANALYSIS; METHODS; EVALUATION; STATISTICS; REACTORS; SAFETY; RISK; ACCIDENTS; RECOMMENDATIONS; MODELS; PIPES; EARTHQUAKES; RADIATION EFFECTS; RADIATION DOSE; CARCINOGENESIS; GENETICS; THYROID; SABOTAGE; CALCULATIONS; EXPOSURE, POPULATION

<513>
Malaro, J.C., U.S. Nuclear Regulatory Commission, Division of Fuel Cycle And Material Safety, Washington, DC.

High-Level Nuclear Waste Management in the United States: A Time for Decisions.

Nuclear Safety 19(3):356-367. (1978, May-June)

The principal public concern associated with the use of nuclear power has been the safe disposal of the wastes involved. Presently an intense, coordinated, and well funded effort to solve the problem is under way. However, the safe management of nuclear wastes not only presents a technological challenge, but also involves significant social, political, and ethical questions. The acceptability of any disposal facilities for high-level radioactive wastes and/or spent fuel which can be built and operated can only be demonstrated through a review and decision process involving all interested parties, including the general public. The mechanism for this exists in the NRC licensing process. Thus it is essential that formal licensing proceedings be initiated and completed at the earliest possible time. Confidence that high-level waste can be disposed of safely is based primarily on current technology capabilities and the size of the federal program. Any serious delays could alter this confidence. (CAJ)

WASTE DISPOSAL; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE; LICENSING; STANDARDS, FEDERAL

<514>
Martin, J.E., U.S. Congress, House of Representatives, Committee on Interior and Insular Affairs.

Statement of Dr. James E. Martin.

Oversight Hearings before the Subcommittee on Energy and the Environment, 95th Congress, First Session, Washington, D.C., May 16, 17, and 20, 1977, Serial No. 95-15, (pp. 227-262), 315 pp. (1977)

A review of the Environmental Protection Agency's (EPA) role in waste management is presented. There are three programs that require satisfactory execution in resolving the problem of radwaste disposal: 1) statement of requirements for protection of the environment and public health; 2) development of technologies, sites and the conduct of operations; and 3) regulation of activities at specific sites to assure overall safety. The first program is the only one EPA is concerned with. The source of the authority in nuclear management is cited. At present, EPA is involved with setting adequate criteria for storage and disposal of high-level radwaste with the intent to use such criteria and the resultant technology for low-level wastes. It will achieve this responsibility by: providing federal radiation guidance; establishing general numerical standards to define present and future conditions; and developing of standards and regulations in areas where EPA has regulatory jurisdiction. General environmental criteria will be set forth. Other activities in waste management are naturally-occurring radioactive wastes,

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REGULATORY AND ECONOMIC ASPECTS

<5 14> CONT.
low-level wastes, transuranium contaminated wastes, decommissioning, and ocean disposal. (NDV)

CONTAINMENT; DECOMMISSIONING; ENVIRONMENT; LEGISLATION; REVIEWS; STANDARDS, FEDERAL; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, TRANSURANIC

<515>
Martin, J.E., U.S. Environmental Protection Agency, Washington, DC.

Environmental Protection Requirements for Radioactive Wastes.

Waste Management and Fuel Cycles '78, R.G. Post and R.E. Wacks (Eds.), Proceedings of a Symposium, Tucson, AZ, March 6-8, 1978. Arizona Board of Regents, (pp. 46-53), 661 pp.(1978, March)

Outlined is the Environmental Protection Agency's responsibilities in developing criteria and specific numerical standards for dealing with radioactive wastes. The authority for EPA to set radiation protection standards was derived from the former Federal Radiation Council and the Reorganization Plan of 1970. The first step in a three step program is the development of pertinent environmental protection criteria for all wastes. The second step is to establish general numerical standards to define present and future conditions that must be met for protection of the environment for high-level wastes. Development of standards and regulations in areas where EPA has specific regulatory authority, such as under the Ocean Dumping Act, is the last step. This program is carried out in conjunction with other agencies. Initial environmental criteria address the following areas: features of radwastes and their hazard potential over time at various level of control; importance of risk estimates; goals of control and the types of institutions engineered, and natural barriers for meeting the goals; approaches for determining allowable levels of short- and long-term risks associated with various methods of disposal for types of wastes; and retrievability, monitoring, and transfer of information. An analysis for high-level waste is being done first. Other EPA activities include naturally-occurring radioactive wastes, low-level wastes, transuranium contaminated wastes, decommissioning, ocean disposal. (NDV)

DISPOSAL SITE; ENVIRONMENT; HAZARD ANALYSIS; LEGISLATION; MONITORING; METHODS; RETRIEVABILITY; SEA DISPOSAL; SITE SELECTION; SITE SURVEILLANCE; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, TRANSURANIC; DECOMMISSIONING

<516>
Mitre Corporation, Nuclear Energy Policy Study Group.

Radioactive Waste.

Nuclear Power Issues and Choices, Chapter 8, (pp. 283-267), 618 pp.

Problems associated with the management of nuclear wastes are discussed. The discussion begins with the nature of nuclear waste as it originates in the generation of nuclear power, indicates the problems inherent in whether the waste is reprocessed or recycled, suggests possible modes of disposal and examines the institutional and programmatic approaches to waste management, both in this country and abroad. Disposal of waste in stable geological formations with little or no contact with the ground water appears to be an adequate solution. Reasonable scenarios of repository failure indicate that the consequences would neither be catastrophic or even substantial. Since the security of the repository is mainly related to its integrity, there is little difference in the risk in storing spent fuel and reprocessed waste. Management of the wastes in the past has lead to problems such as leaky tanks, hence future decisions should be carefully considered. Problems needing attention are: emissions from tailings, releases of Kr 85 from reactors, and improper treatment of transuranic contaminated materials. Since reprocessing increases the complexity and magnitude of waste management and the appropriate regulations are lacking, deferring a decision to reprocess would be best to allow development of the regulations. Flexible temporary storage facilities should be provided until more secure ones are available. The appropriate government agencies in the past have not worked together on waste management, with deleterious effects on public and technical community confidence. Military wastes should be integrated with the commercial waste insofar as possible. If the breeder reactor becomes a reality, then the processing of military wastes would prove useful. (NDV)

BURIAL; CONTAINER INTEGRITY; DISPOSAL SITE; ECONOMICS; FUEL CYCLES; IMMOBILIZATION; REACTORS; RECOMMENDATIONS; SAFETY; SITE EVALUATION; SITE SELECTION; WASTE DISPOSAL; WASTE MANAGEMENT; WASTE STORAGE; WASTES, RADIOACTIVE; NUCLEAR POWER; GEOLOGY; SOCIAL ASPECTS

<517>
Nielsen, S.O., Symplexor Engineering Consultants, Incorporated, Holte, Denmark.

Comparison of Some Geologic and Ocean Disposal Concepts Regarding Realistic Modeling that Allows Objective Risk Assessment to be Made.

CONF-781127; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 549-558), 563 pp.(1979)

A concept concerning risk assessment is discussed. Some guidelines are stated for examining geological disposal concepts and they are: 1) period of isolation time to be minimized by partitioning if possible, 2) several distinct barriers between waste and biosphere, 3) waste form well-characterized, 4) geologic medium must be competent to handle repository, and 5) lead time for pilot project only 5-10 years. Some general criticisms grow out of the guidelines. They are: objective risk assessment is difficult beyond 1000 years, evaluation of the receiving formation is subject to interpretation, monitoring and retrieval are

<517> CONT.
difficult, and lead time may be longer than 10 years. These criticisms are valid for geologic and oceanic disposal models. (NDW)

HAZARD ANALYSIS; SEA DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; MODELS; WASTE RETRIEVAL; PARTITIONING

<518>
Olivier, J.P., Organization for Economic Cooperation and Development, Paris, France.

Sea Disposal of Packaged Radioactive Wastes.

CONF-750811; Nuclear Energy Maturity Proceedings of the 1st European Nuclear Society Conference, Paris, France, April 21-25, 1975, (pp. 693-695). (1975)

This paper summarizes criteria considered by the OECD Nuclear Energy Agency in the sea disposal of radioactive wastes. The limiting capacity of the environment for various waste types for the North Atlantic area, based on a 1977 NEA Hazard Assessment, is presented. The values are 10(E+9) Ci/yr for alpha wastes containing mainly Pu 239, 10(E+6) Ci/yr for alpha wastes containing mainly Ra 226, 10(E+11) Ci/yr for beta-gamma wastes (excluding tritium), 10(E+12) Ci/yr for aged mixed fission products, and 10(E+15) Ci/yr for tritium. Wastes considered unsuitable for dumping are those having a Ci content greater than 10 Ci/MT for alpha wastes with half-life exceeding 50 yrs (but excluding Ra 26 whose overall limit was fixed at 100 Ci/yr per dumping area), those containing more than 1000 Ci/MT of beta-gamma wastes (excluding tritium), but with a 100 Ci/MT limit for Sr 90 and Cs 137, and those containing more than 10(E+6) Ci/MT of tritium. In 1970, the cost of sea disposal was about 12.50 Pounds Sterling/MT of waste, including insurance, ship rental, harbor costs, but not costs of conditioning, land transport, and loading of wastes on the ship. Wastes dumped between 1967 and 1974 account for 34,760 MT, with a total alpha activity of about 3220 Ci and beta-gamma activity (including H3) of 175,000 Ci. Criteria for selection of dumping areas and guidelines for packaging of wastes are summarized. Also discussed briefly are other regulations for control of disposal operations, and the effects of the enforcement of guidelines of the London Convention on the Prevention of Marine Pollution from Dumping of Wastes and Other Matter (1972), which are expected to be minimal. (LKM)

SEA DISPOSAL; RELEASE LIMITS; WASTES, HIGH-LEVEL; COST BENEFIT ANALYSIS; STANDARDS, INTERNATIONAL; WASTE DISPOSAL; ECONOMICS

<519>
Platt, A.M., Battelle-Pacific Northwest Laboratories, Richland, WA.

The Alternatives for Managing Post-Light-Water Reactor Nuclear Waste.

CONF-760310; STI/PUB/433; IAEA-SM-207/67; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 1, (pp. 59-75), 397 pp. (1976)

Management options for nuclear waste from LWR fuel cycles are assessed in terms of commercially available technologies. Two extremes in fuel cycling are considered - one in which waste is discarded and one in which uranium and plutonium are recycled to a maximum. Waste types are categorized and treatment options summarized in flow diagrams. Storage and disposal programs and procedures for site selection, with probabilities for success, are also discussed. The material presented in this paper is handled at much greater depth in ERDA-76-43, the Technical Alternatives Document, issued April 1976. (LKM)

Leaching Rate

WASTE MANAGEMENT; WASTE DISPOSAL; FUEL CYCLES; FUEL REPROCESSING; ECONOMICS; WASTE STORAGE; REVIEWS; THEORETICAL STUDIES; STORAGE, GEOLOGIC

<520>
Rotty, R.M., A.M. Perry, and B.M. Reister, Oak Ridge Associated Universities, Institute for Energy Analysis, Oak Ridge, TN.

Net Energy from Nuclear Power.

IEA-75-3; 131 pp. (1975, November)

An analysis of net energy from nuclear power plants was carried out, taking into consideration energy requirements as they relate to reactor type, concentration of uranium in the ore, enrichment tails assays, and possible recycle of uranium and plutonium. Four reactor types were considered: pressurized water, boiling water, high-temperature gas-cooled, and heavy water (CANDU). The energy requirements of systems employing both conventional (current) ores with uranium concentration of 0.176% and Chattanooga Shales with uranium concentration of 0.006% were determined. Tails assays of 0.36, 0.275%, and 0.2% U 235 have been considered in this study. Data is given for no recycle, uranium recycle only, and uranium plus plutonium recycle. Nuclear power systems were divided into 9 separate components or processes, and separate energy analyses were made for each. Starting with the energy requirements in the mining process, and continuing through fuel reprocessing and waste storage, an evaluation of both electrical energy requirements and thermal energy requirements of each process was made. All of the energy, direct and indirect, required by the processing of uranium to produce electrical power was obtained by adding the quantities for the individual processes. With conventional ores the process requiring the largest single energy expenditure is that of enrichment. Lower concentration ores require larger inputs in mining and milling, and for Chattanooga Shale these requirements dominate the total system. Construction of the power plant is another source of large energy expenditure, and this input is required well in advance of the time that electrical output can be expected from the system. Analysis showed that the energy requirements for the storage of high-level radioactive wastes is small compared to the other energy requirements in the uranium fuel cycle. Direct energy requirements, energy content in materials, and the energy of construction were all considered as in the other processes

<520>

REGULATORY AND ECONOMIC ASPECTS

<520> CONT.
of the fuel cycle. The energy content of process materials was the dominant input; the relatively high values for process material energy requirements can be traced to the need for hydraulic cement and for steel (needed largely for the storage canisters). High-level waste disposal energy requirements for a conceptual plant with a 2000 canisters/yr waste capacity [one 1000 MW(e) reactor required 10 canisters/yr] are as follows: Direct energy requirements = electricity purchase of 870 MWH, and liquid fuel (diesel) requirement of 13,869 X 10(E+6) BTU [100,000 gal at 138,690 BTU/gal]; energy in process materials = 30,309 MWH [1,054,266 X 10(E+6) BTU]; plant construction energy = energy in construction materials 1,967 MWH [82,162 X 10(E+6) BTU], and direct energy in construction 150 MWH [70,000 X 10(E+6) BTU]. Total energy required at the waste disposal plant per year is 33,379 MWH [1,221,131 X 10(E+6) BTU]; energy required per each 1000 MW(e) operating reactor is 167 MWH [6,106 X 10(E+6) BTU]. (LEN)

REACTORS; NUCLEAR POWER; ORES; ENRICHMENT; RECYCLING; FUEL REPROCESSING; REACTORS, PRESSURIZED WATER; REACTORS, BOILING WATER; REACTORS, HIGH-TEMPERATURE GAS-COOLED; REACTORS, CANDU; PLANTS, WASTE TREATMENT; ECONOMICS; THEORETICAL STUDIES; COSTS; WASTE TREATMENT; WASTE STORAGE; WASTES, HIGH-LEVEL

<521>
Sears, M.E., R.E. Blanco, R.C. Finney, G.S. Hill, F.E. Moore, and J.P. Witherspoon, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Conversion of Yellow Cake to Uranium Hexafluoride. Part 1. The Fluorination-Fractionation Process.

ORNL/NUREG/TM-7; 305 pp. (1977, September)

A cost/benefit study was made to determine the cost and effectiveness of radioactive waste treatment systems for decreasing the release of radioactive materials and chemicals from a model uranium hexafluoride (UF6) production plant using the fluorination-fractionation (dry hydrofluor) process, and to evaluate the radiological impact (dose commitment) of the released materials on the environment. This study is designed to assist in defining the term "as low as is reasonably achievable" (ALARA) in relation to limiting the release of radioactive materials from nuclear facilities. The model plant processes 10,000 metric tons of uranium per year. Base-case waste treatment is the minimum necessary to operate the process. Effluents meet the radiological requirements listed in the Code of Federal Regulations, Title 10, Part 20 (10 CFR 20), Appendix B, Table II, but may not be acceptable chemically at all sites. Additional radwaste treatment techniques are applied to the base-case plant in a series of case studies to decrease the amounts of radioactive materials released and to reduce the radiological dose commitment to the population in the surrounding area. The costs for the added waste treatment operations and the corresponding dose commitment are calculated for each case. In the final analysis, radiological dose is plotted vs the annual cost for treatment of the radwastes. The status of the radwaste treatment methods used in the case studies is discussed. Much of the technology used in the advanced cases will require development and demonstration or else is proprietary and unavailable for immediate use. The methodology and assumptions for the radiological doses are found in ORNL-4992. (Auth)

COST BENEFIT ANALYSIS; WASTE TREATMENT; DOSE COMMITMENTS; THEORETICAL STUDIES; EFFLUENTS; EFFLUENTS, AIRBORNE; URANIUM COMPOUNDS; WASTES, RADIOACTIVE; WASTES, COMMERCIAL; FUEL FABRICATION; FUELS; EFFLUENTS, LIQUID; EFFLUENTS, CHEMICAL; MODELS; MODELS, MATHEMATICAL

<522>
Shoup, R.L., Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

International Waste-Management Symposium.

Nuclear Safety 18(4):502-512. (1977, July-August)

An International Symposium on the Management of Wastes from the LWR Fuel Cycle was held in Denver, Colorado, on July 11-16, 1976. The symposium covered a broad range of topics--from policy issues to technology. Presentations were made by national and international speakers involved in all aspects of waste management--government and agency officials; laboratory managers, directors, and researchers; and industrial representatives. Many speakers advocated programatic action on programs for the management of commercial nuclear wastes to complete the light-water reactor (LWR) fuel cycle. The industrialized nations' demand for increasing supplies of energy and their increasing dependence on nuclear energy to fulfill this demand will necessitate the development of an acceptable solution to the disposal of nuclear wastes within the next decade for some industrial nations. Waste-disposal technology should be implemented on a commercial scale, but the commercialization must be accompanied by the decision to use the technology. An important issue is the use of nuclear energy is the question of sharing the technology with the less industrialized nations and with nations that may not have suitable means to dispose of nuclear wastes. The establishment of international and multinational cooperation will be an important key in realizing this objective. Pressing issues that international organizations or task groups will have to address are ocean disposal, plutonium recycling and safeguards, and disposal criteria. The importance of achieving a viable waste-management program is made evident by the increased funding and attention that the back end of the fuel cycle is now receiving. (Auth)

BURIAL; CONTAINMENT; COST BENEFIT ANALYSIS; DISPOSAL SITE; DILUTION; DIFFUSION; ENVIRONMENT; FUEL ELEMENTS; GEOLOGY; HAZARDS; HYDROLOGY; PREDICTIONS; SAFETY; STANDARDS, FEDERAL; WASTE DISPOSAL; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, SOLID; WASTES, TRANSURANIC; SOLIDIFICATION

<523>
Sunder Rajan, R.S., V. Ravindranath, R.S. Kanta, P.K. Sinha Ray, and K.T. Thomas, Bhabha Atomic Research Centre, Trombay, Bombay, India.

Long-Term Planning for Management of Aqueous Wastes from Fuel Reprocessing Plants.

CONF-760310; STI/PUB/433; IAEA-SM-207/87; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 1, (pp. 15-22), 397 pp. (1976)

This paper reviews Indian policy regarding management of high-and intermediate-level liquid wastes from fuel reprocessing plants, and attempts to estimate the generation of such wastes over the next 25 years. Methods for treatment and disposal of wastes are discussed, including interim storage, incorporation of waste salts in bitumen, solidification of high-level wastes, and ultimately geologic storage/disposal. As reported in an earlier paper, experimental solidification of high-level waste oxides in borosilicate matrices has yielded products with melting ranges of 1000 to 1100 degrees C and leach rates of 10(E-5) to 10(E-6) g cm (E-2)d (E-1). Waste oxide incorporation is approximately 25%. Long term studies of the behavior of these melts under intense radiation and thermal stresses are ongoing. Another major consideration has been the management of actinide-bearing wastes. Research is continuing on separation of actinides before waste solidification, or, alternatively, conversion of actinide-bearing wastes into insoluble alumino-silicates. (LKS)

BITUMINIZATION; CONCENTRATION; CONTAINMENT; CORROSION; FUEL REPROCESSING; PREDICTIONS; REACTORS, BREEDER; REPROCESSING; SALTS; SOLIDIFICATION; STORAGE, GEOLOGIC; WASTE STORAGE; WASTE MANAGEMENT; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, SOLID; ACTINIDES; THORIUM; URANIUM; PLUTONIUM

<524>
U.S. Congress, House of Representatives, Committee on Interior and Insular Affairs.

Nuclear Waste Management.

Oversight Hearings before the Subcommittee on Energy and the Environment, 95th Congress, First Session, Washington, D.C., May 16, 17, and 20, 1977, Serial No. 95-15, 315 pp. (1977)

The hearings on waste management focus on the public concern of the various wastes produced by nuclear industry and their disposal. Five different waste streams are considered: 1) low-level wastes either contaminated with fission products only or with transuranics (TRU); 2) high-level processing wastes; 3) spent fuel; 4) uranium mill tailings; and 5) gaseous effluents. Various disposal methods and needs for an overall policy are discussed. Sections relevant to the data base are in separate abstracts. (NDV)

COMMUNITIES; FISSION PRODUCTS; HAZARDS; LEGISLATION; NUCLEAR POWER; REVIEWS; STANDARDS, FEDERAL; STANDARDS, INTERNATIONAL; STANDARDS, STATE; WASTE DISPOSAL; WASTE MANAGEMENT; WASTE STORAGE; WASTE TREATMENT; WASTES, GASEOUS; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, SOLID; WASTES, TRANSURANIC

<525>
U.S. Department of Energy, Office of Nuclear Waste Management, Washington, DC.

Costs.

DOE/ET-0029; Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 15, (pp. 15-1 - 15-4), 586 pp. (1979, February)

Cost estimates for the alternatives and concepts studied are based on 1978 dollars. Included are capital costs, operating and maintenance costs, and costs for D and D of facilities. Some of the indirect capital costs, such as site characterization, and environmental preparation, are not included. Where applicable, the costs of shipping the waste to the Federal Repository and of long-term management of the waste at the Federal Repository are included. The estimates have large associated uncertainties. The lowest costs would be associated with Alternatives 1 and 2 (leave TRU waste as is, and leave TRU waste as is with improved confinement), ranging from $60 million to $71 million. The cost to retrieve and process the waste and to ship to the Federal Repository and dispose of it there would range from $371 million to $389 million (Alternatives 3, 4, and 6). The cost of retrieval, processing, and onsite disposal would range from $220 million to $567 million, depending on the methods of processing and of disposal. (Auth) (RAF)

COST BENEFIT ANALYSIS; ECONOMICS; WASTE MANAGEMENT; WASTES, TRANSURANIC; TRANSPORTATION; REPOSITORY; WASTE PROCESSING; RETRIEVABILITY; WASTE DISPOSAL

<526>
U.S. Environmental Protection Agency, Office of Radiation Programs, Waste Environmental Standards Program, Washington, D.C..

Considerations of Environmental Protection Criteria for Radioactive Waste.

U.S. EPA Background Report; 61 pp. (1978, February)

Five areas are addressed which are to be used in the development of environmental standards for major classes of radioactive waste. Environmental protection requirements to be developed are: 1) features of radioactive materials that require them to be designated as radioactive wastes and their hazard potential over time and at various levels of controls; 2) risk estimates in arriving at levels of control and the factors that should be reflected in such determinations; 3) the goals of control and the types of institutional, engineered, and natural-barrier controls for meeting such goals; 4) the approaches for determining the allowable levels of short- and long-term risks associated with various means of disposal for types of waste materials; 5) other considerations for environmental protection such as retrievability, monitoring, and transfer of information to

<526>

REGULATORY AND ECONOMIC ASPECTS

<526> CONT.
succeeding generations. Public input was received from two public workshops in February and April 1977 to address relevant policy and technical issues. Consideration was given to the views expressed in these workshops in drafting this document and in the initial formulations of environmental criteria. (PAF)

Major findings of the workshops are summarized in an appendix.

WASTES, RADIOACTIVE; STANDARDS, FEDERAL; HAZARD ANALYSIS; RADIATION HAZARDS; WASTE DISPOSAL; RETRIEVABILITY; MONITORING; INFORMATION; MAN; WASTES, HIGH-LEVEL; RADIATION, GAMMA; FISSION PRODUCTS; ENVIRONMENTAL EXPOSURE PATHWAY; WASTES, LOW-LEVEL; WASTES, TRANSURANIC; ALPHA PARTICLES; ECONOMICS; CONTAINMENT

<527>
Effen, R.J., Queen's University, Kingston, Ontario, Canada.

Let's Go Slowly on a Nuclear Power Program Until We've Solved Waste Problems.

Science Forum 59:3-8.(1977, October)

A comprehensive discussion of the dilemma facing the nuclear industry on the topics of expansion and reprocessing is presented. In order to convince the public, governments, and regulatory agencies that the proposed methods are sound, a demonstration site should be set. Permanent disposal of highly radioactive wastes would best be done by vitrification. Guidelines similar to those of low-level wastes should be developed for high-level wastes. Until the reprocessing issue is decided the industry will have to proceed at a slower pace. Although the problems are immense in magnitude they are solvable. (NDW)

ACCIDENTS; ENVIRONMENT; FUEL REPROCESSING; LEGISLATION; PLANTS, WASTE TREATMENT; REPOSITORY; POPULATIONS; STANDARDS, FEDERAL; WASTE MANAGEMENT; WASTES, RADIOACTIVE

<528>
Zerby, C.D., and W.C. McClain, Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

Waste Isolation in Geologic Formations in the United States of America.

LONI-7; CONF-760701; Management of Waste from the LWR Fuel Cycle, Proceedings of an International Symposium, Denver, CO, July 11-16, 1976, (21 pp.).(1976)

The Energy Research and Development Administration (ERDA) program for the establishment of terminal storage facilities for commercial radioactive wastes in deep geologic formations was recently reorganized as the National Waste Terminal Storage (NWTS) program under the direction of Union Carbide Corporation - Nuclear Division. General plans for implementing this expanded program call for geologic investigations and feasibility confirmation studies at multiple geographic locations, leading to pilot plant construction and operation with possible future conversion into a Federal Repository. The pilot plant operations will be experimental facilities having limited capacity to store actual waste in a readily retrievable configuration. The first two pilot plants are planned to start operations simultaneously in the mid-1980's. Geologic investigations are now in progress or planned in study areas of the interior basins of the Gulf Coast Salt Dome Province, in the Salina Salt basin and in the Paradox Basin in an effort to identify acceptable locations for these initial facilities. Subsequent pilot plants will be located in other formations. Preliminary geologic evaluations have been initiated in the Paleozoic shales and limestones, Triassic shale basins along the east coast, Mesozoic shales of the Gulf Coast and northern high plains and certain crystalline igneous rocks. Most of the required engineering testing of disposal in salt formations has been completed in previous programs. However, the establishment of pilot plants in the other rock types will require a sequence of in situ testing designed to develop the information necessary to both demonstrate the feasibility of waste disposal in that particular formation and provide the data for facility designs. (Auth)

GEOLOGIC FORMATIONS; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; REPOSITORY; RETRIEVABILITY; SALT DOMES; SHALES; LIMESTONES; IGNEOUS ROCKS; WASTE DISPOSAL; WIPP; SITE SELECTION

<529>
Roch, A.L., and R.E. Kibbe, Union Carbide Corporation, Nuclear Division, Office of Waste Isolation, Oak Ridge, TN.

A Waste Repository in a Geologic Formation.

Y/OWI/TM-52; CONF-780622; American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (pp. 82-83); Transactions of the American Nuclear Society 28:82-83.(1978, February 6; 1978, June)

A typical salt waste repository design under the current DOE National Waste Terminal Storage (NWTS) program would have 150-200 acres of surface facilities and about 2000 acres of subterranean waste emplacement horizon. The surface facilities are characterized by three major structures: a canistered waste handling building featuring shielded and remote handling facilities, a low-level waste handling building designed for contact handling, and a mine operations building. Each building would have a dedicated mine shaft connecting it to the underground operations. The current NWTS repository design assumes all wastes arrive onsite in canisters. Studies since 1976 have shown that spent fuel canisters can be disposed of using the same basic repository design, with [illegible], modifications for greater canister length (16 ft versus 10 ft for high-level, cladding, and intermediate-level wastes) and higher decay heat (accumulation; reduction in areal thermal loading by about 50%). Subterranean transport of high-level waste or spent fuel assemblies would be via a low-speed vehicle carrying a shielded cask. Waste would be lowered from the cask into a hole in the floor of the excavated room. Retrievability could be allowed for by placement of a concrete shielding plug over the waste hole; otherwise the hole would be filled with crushed salt to a depth of about 7 ft. Palletized drums of low-level waste would be transported by fork lift and stacked in open rooms. If retrievability is no longer desired, all rooms would be back-filled with salt to 50% of the theoretical salt density before decommissioning. (LEN)

REPOSITORY; DESIGN; SALT DEPOSITS; STORAGE, GEOLOGIC; CANISTERS; SPENT FUELS; WASTES, HIGH-LEVEL; CLADDING; WASTES, INTERMEDIATE-LEVEL; RETRIEVABILITY; WASTES, ICW-LEVEL; DECOMMISSIONING; WASTE DISPOSAL; WASTE STORAGE; WASTE MANAGEMENT; THEORETICAL STUDIES

<530>
Boegly, W.J., Jr., and W.F. Schaffer, Oak Ridge National Laboratory, Oak Ridge, TN.

Hazards Analysis.

ORNL-4555 (Draft); ORNL-CF-70-7-82; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 9, (pp. 9.1-9.69), 822 pp.(1971, April)

Detailed hazard analyses are made on cask design, shipping hazards, topside transfer, fission product containment, and accidents (prevention and types). Irradiated reactor fuel assemblies were used in the demonstration rather than calcined waste or separated fission product sources. Selection of HTR fuel was considered the best overall choice. (CHJ)

Graphs, equations, tables, diagrams, and reference are given.

WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; MINES; DISPOSAL SITE; DESIGN; EQUIPMENT; FIELD STUDIES; EVALUATION; INSTRUMENTS; CASKS; CANISTERS; SAFETY; ACCIDENTS; TRANSPORTATION; SITE EVALUATION; WASTES, SOLID; HAZARD ANALYSIS; EQUATIONS

<531>
Boegly, W.J., Jr., W.F. Schaffer, and R.L. Bradshaw, Oak Ridge National Laboratory, Oak Ridge, TN.

Design and Fabrication of Equipment.

ORNL-4555 (Draft); ORNL-CF-70-7-82; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 7, (pp. 7.1-7.69), 822 pp.(1971, April)

Special equipment needed for major operations in Project Salt Vault are described. The major operations requiring special equipment are: 1) encapsulation of fuel assemblies; 2) modification of a cask for cross-country shipment; 3) waste charging at the facility/surface; 4) underground transfer, 5) storage in salt; 6) heated-pillar experiment; and 7) instrumentation.

Diagrams, schematic drawings, photos, tables, and references are included.

STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; MINES; DISPOSAL SITE; DESIGN; EVALUATION; EQUIPMENT; CASKS; CANISTERS; FIELD STUDIES; WASTES, SOLID; ENCAPSULATION; TRANSPORTATION

<532>
Boyd, R.D., Sandia Laboratories, Albuquerque, NM.

Forced-Air Cooling of a Waste Isolation Pilot Plant Mine Drift.

SAND-78-1122; 39 pp.(1978, July)

A one dimensional (1-D) model has been formulated as a first attempt at providing WIPP designers with an approximate analysis of the forced air cooling of a WIPP mine drift. This 1-D analysis compares favorably with a 2-D model developed by Science Applications, Inc. At very small time values (weeks), the 1-D predictions are about 40% lower than the 2-D predictions; at 6 months to about 2 years, agreement between the 1-D and 2-D predictions is good (better than 10%). At times greater than 2 years, the 1-D predictions are about 25% higher than the 2-D. One dimensional predictions were made for a repository loading of 75 kW/acre and an inlet air velocity and temperature of 2.57 ft/sec and 92 F, respectively. Results indicate that the time required to cool the floor of a salt mine drift which has had radioactive waste emplaced for more than about 15 years is independent of the time at which ventilation begins. If velocity is significantly increased, cooling time appears to decrease accordingly; however, for the typical WIPP design specifications, cooling

<532>

REPOSITORY DESIGN AND ENGINEERING

<532> CONT.

time is not substantially affected by changes in the inlet flow rate above 70,000 cfm. If the mine surface is rough, i.e., the mean heat transfer coefficient (h) is greater than or equal to 1.3 Btu/hr-sq ft-F, it is estimated that, depending on the time at which cooling is begun, less than one year will be required to cool the salt mine floor to 120 F. However, if the mine surface is smooth, i.e., h is less than or equal to 0.3 Btu/hr-sq ft-F, it is estimated that less than 10 years will be required for cooling to 120 F. It is obvious that before reliable estimates of cooling time can be made, the roughness of the mine surfaces must be determined. Isolated heat transfer measurements are necessary to verify these computed values using the measured roughnesses and the available correlations for rough surfaces. Based on existing correlations for sand roughnesses in pipes, if the roughness of the salt surface is at least 1/8 inch, it appears that the resulting heat transfer coefficient will be greater than 1.0 Btu/hr-sq ft-F (approximately equal to 1.4 Btu/hr-sq ft- for Reynolds number = 2 X 10(E+5)). This i ies that predictions made for h = 1.0 Btu/hr-sq ft-F are conservative, using the typical WIPP design specifications. (Auth) (LKN)

WIPP; MODELS; MODELS, MATHEMATICAL; PREDICTIONS; SALT DEPOSITS; REPOSITORY; WASTE STORAGE; STORAGE, GEOLOGIC; VENTILATION; HEAT TRANSFER; THEORETICAL STUDIES

<533>

Bradshaw, R.L., Oak Ridge National Laboratory, Health Physics Division, Oak Ridge, TN.

Preliminary Design of Experiment.

ORNL-4555 (Draft); ORNL-CF-70-7-82; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 3, (pp. 3.1-3.32), 822 pp.(1971, April)

This chapter concerns the engineering and scientific objectives of the demonstration, which were to determine the requirements for safe and economic disposal in salt formations. Included are discussions on: selection of mine and radiation source, calculation of fuel assembly temperatures, salt dose calculations, experimental layout in the mine, salt temperature patterns in arrays, and experimental instrumentation. Salt flow data were obtained which would aid in the design of an actual disposal facility. (CAJ)

Graphs, schematic drawings and isotherm charts are included.

WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; GEOPHYSICAL SURVEYS; DISPOSAL SITE; DESIGN; THERMAL PROPERTIES; MINES; LABORATORY STUDIES; SITE EVALUATION; WASTES, SOLID; DOSE RATE

<534>

Borleigh, R.W., R.P. Binnall, A.O. DuBois, D.U. Norgren, and A.R. Ortiz, Swedish Nuclear Fuel Supply Company, Stockholm, Sweden; Lawrence Berkeley Laboratory, Earth Sciences Division, Berkeley, CA.

Electrical Heaters for Thermo-Mechanical Tests at the Stripa Mine.

LBL-7063; 52 pp.(1979, January)

Electrical heaters were installed at the Stripa mine in Sweden to simulate the heat flux expected from canisters containing nuclear waste. Three heater types were designed and fabricated: two full scale heaters, 2.6 m in length and 328 mm in diameter, supplying a maximum power output of 5 kW; eight peripheral heaters of 25 mm diameter, supplying 1.1 kW; and eight time scale heaters, one-third the size and power of the full scale heaters. Power can be monitored by panel meters as well as by a computer-based data acquisition system. Both the controller and the heater were designed with a high degree of redundancy in case of component failure. Auxiliary items were provided with the heaters to monitor borehole decrepitation and heater temperature, and to dewater the heater holes. Installation was relatively eventful. The time-scale heater experiment was turned on June 1, 1978. The first full-scale heater was energized on July 3, and the second on August 24. The time-scale heaters have operated over 36,000 hrs and the first of the full-scale heaters has operated 2800 hrs without failures or serious problems. (Auth) (LKN)

HEAT SOURCES; EQUIPMENT; INSTRUMENTS; REPOSITORY; MONITORING; FIELD STUDIES; DESIGN

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Carlsson, H., University of Lulea, Division of Rock Mechanics, Lulea, Sweden.

A Pilot Heater Test in the Stripa Granite.

LBL-7086; SAC-36; 146 pp.(1978, August)

A pilot heater test was carried out at the Stripa mine in central Sweden at a level of 348 m. The type of rock is a granite with a high frequency of fractures. A central main heater with a length of 3 m, a diameter of 30 cm and a total power of 6 kW was placed at the bottom of a 10 m deep borehole. At different radial distances, varying from .85 m to 2.95 m from the heater, stress and temperature changes were monitored. Heating of the rock lasted for 69 days, then the power was turned off to monitor the cooling effects of the rocks. The measured temperature distribution compared fairly well with predicted values. A maximum temperature of 333.9 C was measured on the heater just before it was turned off. The maximum temperature in the rock, as measured 0.85 m from the axis of the heater, was 10.27 C. The thermal isotropy was affected very little by fractures in the rock. By using data from the cooling period, the thermal conductivity of the rock mass was calculated to lambda = 8.8 w/m degree C. The thermally induced stress in the rock mass did not correspond well with predicted values. The predicted stresses were much higher than the measured, normally by a factor of 3 to 8. A stress anisotropy was found to be prominent close to the heater. Measurements in boreholes of the in situ modules were found to be about half of the laboratory determinations. Displacements of major fractures on the floor of the test drift were very small. Water in

<535> CONT.
flow was lower for the duration of the heater test. (Auth) (RAF)

HEAT SOURCES; INSTRUMENTS; GRANITES; ROCKS; STRESS; MECHANICAL PROPERTIES; PHYSICAL PROPERTIES; THERMAL PROPERTIES; FRACTURES; BOREHOLES; ISOTROPY; ANISOTROPY; GROUND WATER; TIME FACTOR; SITE SELECTION; DENSITY; POROSITY; PERMEABILITY; FIELD STUDIES

<536>
D'Appolonia Consulting Engineers, Inc.

The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive Waste.

ONWI-15; 281 pp.(1979, January)

The report reviews and evaluates penetration sealing (borehole plugging and shaft sealing) activities and research which have been conducted in the United States and Europe. An additional objective is to identify important technical considerations for repository sealing. Based on activities and research conducted to date and on specified technical considerations for repository sealing, the report identifies deficiencies in technology and/or data and recommends areas of required research. Although significant knowledge exists relative to materials and techniques for sealing shafts and plugging boreholes (mainly for purposes not related to nuclear waste disposal) it is evident that this knowledge is of limited value in meeting all of the requirements for nuclear repository sealing. Topical areas where improved information is needed include: 1) Site-characterization studies; 2) studies to characterize boreholes and shafts; 3) development of consequence models; 4) development of analytical models; 5) development of design criteria and specifications for plug or seal properties; 6) detailed laboratory, analytical, and field studies; and 7) instrumentation studies. (RAF)

A complete bibliography and list of contacts are given in the appendices.

REVIEWS; BOREHOLES; SHAFTS; SEAL MATERIALS; SEALING; STORAGE, GEOLOGIC; WASTES, RADIOACTIVE; REPOSITORY; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; MECHANICAL PROPERTIES; THERMAL PROPERTIES; ROCKS; SITE SELECTION; DRILLING; LOGGING; GROUND WATER; MODELS; DESIGN; FRACTURES; GROUTING; FIELD STUDIES; LABORATORY STUDIES; PERMEABILITY; MINING; CEMENTS; CONCRETES; WELLS, INJECTION; CAVES; GRAVELS; CLAYS; GRANITES; INSTRUMENTS; SALT DEPOSITS; DISSOLUTION; DIFFUSION; AQUIFERS; PRESSURE EFFECTS; MONITORING; MOISTURE

<537>
Dennis, A.W., J.A. Hilley, L.W. Scully, R.C. Shefelbine, R.E. Stinebough, and W.E. Nowak, Sandia Laboratories, Nuclear Waste Engineering Division, Albuquerque, NM.

Revised Concept for the Waste Isolation Pilot Plant.

SAND-78-1829; 88 pp.(1978, July)

The quantities of remotely-handled waste that must be handled at the Waste Isolation Pilot Plant (WIPP) have been significantly reduced from earlier estimates [from 250 X 10(E+3) cu ft/yr to 10 X 10(E+3) cu ft/yr]. The reduction in the quantity of remote-handled waste has resulted in the following facility design changes: The extent of the remote-handled storage area (the lower horizon in the underground development) has been substantially reduced. This horizon will be used to perform experiments with high-level waste and to store remotely-handled defense transuranic wastes. The separate surface facility and its associated shaft for receiving remote-handled waste have been eliminated. Because of the reduced ventilation requirements, separate shafts for the construction air exhaust and the waste storage air exhaust have been eliminated. Separate buildings for the waste storage air filter facility, the suspect waste treatment facility and the hoist house have been eliminated and their functions have been incorporated into the Waste Handling Building. (RAF)

WIPP; WASTES, TRANSURANIC; SALT DEPOSITS; WASTES, HIGH-LEVEL; DESIGN; REMOTE HANDLING; WASTE STORAGE; VENTILATION; WASTE VOLUME; COSTS

<538>
DiNunno, J.J., NUS Corporation, Rockville, MD.

Environmental Aspects of High-Level Radioactive Waste Disposal in Deep Geologic Formations.

Transactions of the American Nuclear Society 28:81-82; CONF-780622; American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (pp. 81-82).(1978)

Development of a high-level radioactive waste repository will result in environmental impacts depending upon the specific uses to be made of the repository, its design features, and the natural setting. Potential environmental impacts in the near-term are related to: conflicts with lands currently dedicated to public uses or projected for such uses; establishing or improving access to the site; increased traffic due to construction and operation of the site; radiation exposures to people on and off site during transport and storage of radioactive waste materials; air and waterborne effluents during construction and operation; waste rock disposal; off site noise levels during construction and operation; loss or degradation of wildlife habitats; economic, social, and cultural changes in the vicinity; and scenic alterations. Predictable long-term impacts include: alteration of surface elevations over the repository due to thermal expansion, contraction, and subsidence of the overburden; and a slight warming of the formations and any ground waters overlying the host rock due to heat generated by the waste. Random or inadvertant long-term potential impacts include: rapid erosion of strata over the repository; host rock dissolution (in the case of salt) due to water intrusion; geologic faulting; meteorite impact; and inadvertant human penetration of the repository by drilling, etc. The difficulty in making any assessment of these potential impacts is the unknown nature of future population distributions and habits; one approach is to consider a range of impacts using current demographic data and imposing a

<538> CONT.
range of alternative future human conditions. (LKM)

ENVIRONMENT; ENVIRONMENTAL IMPACTS; REPOSITORY; WASTE MANAGEMENT; LAND USE; EXPOSURE, POPULATION; WASTE STORAGE; STORAGE, GEOLOGIC; WASTES, HIGH-LEVEL; PREDICTIONS

<539>
Fossen, J.H., and W.C. McClain, Oak Ridge National Laboratory, Oak Ridge, TN.

Installation of Experimental Equipment.

ORNL-4555 (Draft); ORNL-CF-70-7-42; Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 8, (pp. 8.1-9.19), 822 pp. (1971, April)

A discussion of the installation and check out of the experimental equipment is presented, bringing the preparation for the experiment to a close. The waste charging facility, underground transporter, power distribution system, array hole liners, pillar heaters, and instrumentation are described. Photographs of some of the equipment after installation are included. (CAJ)

Several photographs are included.

WASTE DISPOSAL; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; MINES; DISPOSAL SITE; DESIGN; EQUIPMENT; FIELD STUDIES; EVALUATION; INSTRUMENTS; WASTES, SOLID

<540>
Fossum, A.F., RE/SPEC, Inc., Rapid City, SD.

Structural Analysis of Salt Cavities Formed by Solution Mining: 1. Method of Analysis and Preliminary Results for Spherical Cavities.

ORNL/SUB-4269/19; Technical Memorandum Report RSI-0043; 30 pp. (1976, January, 15)

This report presents a description of analysis procedures and preliminary results for a simulation of the structural stability of cavities formed by solution mining in salt domes. Primary emphasis is placed on the methodology of the finite element analysis, and the results of preliminary calculations for a spherically-shaped cavity. In particular, the effects of depth (i.e., initial state of in situ stress), shape, volume (i.e., physical dimensions of the cavity), and sequence of salt excavation/fluid evacuation on the timewise structural stability of a cavity are considered. The cavity shape is assumed to be axisymmetric and the salt is assumed to exhibit nonlinear creep behavior. The computer code used for this analysis is RSI/SCAMP (Solution Cavity Analysis with Mining Procedures), derived from certain aspects of the thermo/viscoelastic code RSI/TEVCO. The cavity is situated at a depth of 2000 ft, with a diameter of 300 ft. The salt is assumed to have a modulus of elasticity of 1.5 X 10(E+6) psi and a Poisson's Ratio of 0.35. The cavity is assumed to fill with water under hydrostatic head immediately after excavation, and is allowed to creep for 32 hrs before the water is evacuated and creep continues. The preliminary results of the analysis establish that gradual construction effects, i.e. sequential excavation, may be significant in the evolution of the near field stresses and displacements in the cavity wall. The influence of the cavity on the country salt is of a relatively local nature, with most of the perturbation occurring within one radius of the cavity wall. Most of the stress relaxation occurs very early, essentially in the first day following an excavation. Vertical stresses after 10,000 hrs are essentially the same as those 56 hrs after the water evacuation. Displacement of the cavity wall increases with time at a decreasing rate for the state of initial premining stress and the rheological law considered in this study. For a spherical cavity excavated in a salt mass with an initial premining stress state corresponding to a gravitating elastic medium, the vertical closure of the cavity is greater than the horizontal closure. (Auth) (LKM)

STRUCTURAL STABILITY; SALT DEPOSITS; WASTE STORAGE; STORAGE, GEOLOGIC; WASTE DISPOSAL; MODELS; MODELS, MATHEMATICAL; COMPUTER PROGRAMS; COMPUTER CODES; CAVITIES; MINING; DISSOLUTION; SALT DOMES; STRESS; CREEP; GROUND WATER; EXCAVATION; THEORETICAL STUDIES

<541>
Grams, W.H., RE/SPEC, Inc., Rapid City, SD.

Design Aspects of the Alpha Repository: 2. Conceptual Layouts of Underground Storage Facilities.

ORNL/SUB-4269/9; Technical Memorandum Report RSI-0028; 74 pp. (1975, June 30)

Five conceptual repository layouts were developed provide a composite view of the various aspects of repository designs, and to identify in a quantitative fashion the advantages or disadvantages of certain geometrical patterns of panel development. The following layouts were developed: 1) linear repository layout (one panel); 2) bow-tie repository layout (two panels); 3) Maltese Cross repository layout (four panels); 4) snowflake repository layout (five panels); and 5) sash window repository layout (eight panels). In order to compare the relative merits of the five layouts, consideration was given to excavation/haulage requirements, ventilation design, safety features, and capability for expansion. The snowflake repository had the most favorable rating on an overall basis, and the linear repository layout the least favorable rating. The highest rating of the snowflake repository was based on the following characteristics: 1) the close proximity of the storage rooms to the main hoisting facilities; 2) the minimum amount of development work required prior to the initiation of waste emplacement; 3) favorable ventilation design; and 4) favorable safety features. (RAF)

Extensive illustrations of conceptual repository layouts are given.

REPOSITORY; ALPHA PARTICLES; STORAGE, GEOLOGIC; DESIGN; SALT DOMES; EXCAVATION; SAFETY; WASTES, SOLID; CLADDING; WASTE VOLUME; CONTAINERS; CANISTERS; VENTILATION; THEORETICAL STUDIES

<542>
Greenberg, J., W.K. Winegardner, P.J. Pelto, J.W. Voss, J.A. Stottlemyre, I.A. Forbes, J.B. Fussell, and E.C. Burkholder, Battelle-Pacific Northwest Laboratories, Richland, WA; Energy Research Group, Incorporated, Waltham, MA; University of Tennessee, Knoxville, TN.

Program Scenario Analysis Methods for Use in Assessing the Safety of the Geologic Isolation of Nuclear Waste.

PNL-2643; 42 pp.(1978, November)

In order to analyze the potential modes for breach of a nuclear waste repository, an appropriate methodology for analysis must be chosen. Several possible approaches were selected for analysis and evaluated. The method should: 1) be quantitative and scientifically based; 2) model the potential disruptive events and processes; 3) model the system through time to the failure event in sufficient detail to provide input to a subsequent consequence analysis; and 4) be compatible with the currently foreseeable level-of-knowledge of system data and understanding of the operative events and processes. These criteria were applied to the following methodologies: expert opinion, perspectives analysis, fault trees and/or event trees, classical systems analysis, Markov chains, and Monte Carlo simulation. Expert opinion, fault trees/event trees, Monte Carlo simulation and classical systems analysis were judged to have the greatest potential application to the problem of scenario analysis. The methods were found to be constrained by limited data and by the knowledge of the processes governing the system--some more than others. It was determined that no single method is clearly superior to others when measured against all the criteria. Therefore, to get the best understanding of system behavior, a combination of the methods is recommended. Monte Carlo simulation was judged to be the most suitable matrix in which to incorporate a combination of methods. (RAP)

STORAGE, GEOLOGIC; SAFETY; WASTES, RADIOACTIVE; REPOSITORY; HAZARD ANALYSIS; THEORETICAL STUDIES; MODELS; METHODS; COMPUTER PROGRAMS

<543>
Koplik, C.M., D.L. Pentz, and R. Talbot, The Analytic Sciences Corp., Reading, MA.

Information Base for Waste Repository Design, Volume 6. Excavation Technology.

NUREG/CR-0495; 76 pp.(1979, March)

This report discusses the excavation technology to be used for a nuclear waste facility in a deep geologic formation. Particular emphasis is placed on the requirement to minimize the disturbance to the repository rock. This can reduce the probability of radionuclide migration through the rock surrounding the excavation. An information base is provided to assist in the evaluation of mining methods. Features of particular interest discussed are: Mechanical ripping or boring excavation methods cause much less damage to the surrounding rock than the drill-blast method. However, the drill-blast method may be the only effective technique for excavation of large openings in hard rock (basalt, granite, etc.). The disturbance to the rock from the drill-blast method can be minimized by using contour blasting. Conventional boring excavation methods may be applicable for use in a repository in hard rock rather than drill-blast methods. However, these tunnels borers are likely to be very large, costly, and inflexible in operation. Repository design would have to be modified for their use. A methodology is presented which can be used to determine performance requirements for excavation techniques. The following key points are noted: For excavations made with continuous mining or boring machines, damage to the surrounding rock will be minimal. Therefore, performance criteria will be primarily concerned with excavation support procedures. For excavations made with drill-blast methods, fracturing of the surrounding rock can be severe. Strict criteria on blasting procedures will be required. The perimeter of the excavation must be shaped when the drill-blast method is adopted to reduce rock disturbance. (Auth) (LKH)

DESIGN; METHODS; EXCAVATION; REPOSITORY; STORAGE, GEOLOGIC; DRILLING; FRACTURES; EXPLOSIONS, NON-NUCLEAR; MINING; EQUIPMENT; STABILIZATION, PHYSICAL; WASTE DISPOSAL; WASTE STORAGE; WASTES, RADIOACTIVE; ROCKS

<544>
Lenchman, M.S., and W.P. Bishop, U.S. Nuclear Regulatory Commission, Washington, DC.

Relevance of Natural Retention Experience to Nuclear Waste Management.

IAEA-TC-119/30; Natural Fission Reactors, Proceedings of an IAEA Symposium, Paris, France, December 19, 1977. International Atomic Energy Agency, Vienna, (pp. 700-713).(1978)

A current need exists for providing the public and its representatives with enough information to judge the effectiveness and safety of nuclear waste disposal. This need might be largely fulfilled by careful and understandable presentations of relevant information from the experiences of retaining Oklo wastes and other radioactive materials in inadvertant repositories in geologic media. Any reliance on the proven performances of natural and inadvertant experiences with wastes requires that man-made wastes mimic those of nature, both in their composition and in their surrounding environment. For this reason, it is important to develop methods for confidently mimicking nature and, if successful, to convincingly present these to the public and government officials for their evaluation. The following information seems relevant for waste management: 1) presentation of the quantities and locations of radioactive wastes in terms that the public can qualitatively evaluate; 2) presentation of the alternative forms for solidified wastes, expressed in terms that provide the public with a feeling for retention performances over different time spans; 3) presentation of the hazards involved in fabricating and handling the different solidified wastes, both to the public and occupationally; and 4) assurance, to the extent that scientists and engineers can provide, about how well various waste management practices would mimic the successes or failures of waste retention in natural and inadvertent repositories. Even less-than-convincing presentation can nevertheless provide openness of information

<504>

REPOSITORY DESIGN AND ENGINEERING

<504> CONT.
for the public. If, for various reasons, deliberate geological repositories are rejected, interim storage would need to be extended. The Oklo phenomenon would then provide limited relevant experience. Permanent disposal ultimately might be by some other means, such as seabed or extraterrestrial, in which case no previous experience of any significance appears to be relevant and available. (Auth) (LKW)

REACTORS, NATURAL; SOCIAL ASPECTS; RADIONUCLIDE MIGRATION; WASTE STORAGE; WASTE DISPOSAL; STORAGE, GEOLOGIC; WASTE MANAGEMENT; WASTE-ROCK INTERACTIONS; THEORETICAL STUDIES; WASTE TREATMENT

<505>
Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume B. Repository Preconceptual Design Studies: Salt.

Y/OWI/TM-36/B; 88 pp. (1978, April)

This report describes in detail a preconceptual design for a nuclear waste storage facility in salt. The facility design consists of several chambers excavated deep within a geologic formation together with access shafts and supportive surface structures. Surface facilities and equipment are outlined, buildings are described, and detailed sequences of waste handling operations are given. Descriptions of facility power sources, electric services, utilization voltages, wiring, and sanitary waste disposal systems are provided. The surface facilities are designed to provide the following functions: receipt and unloading of wastes in specially designed containers from rail cars; transfer of containers to inspection stations; decontamination of containers, if required; overpacking when needed; and preparation of waste for descent to the storage level. There are two basic waste handling streams: one for wastes requiring shielding and hot-cell facilities (canistered waste), and one for wastes that do not (low-level waste). Five shafts are provided in the facility design: a canistered waste shaft to the storage level; a man and material shaft, also used for transporting excavated material to the surface; a ventilation supply shaft; and a ventilation exhaust shaft. Shaft boring and lining plans are described. All high-level wastes from LWR fuel cycles, PWR and BWR spent fuel assemblies, and intermediate-level TRU and cladding wastes will arrive at the repository in steel canisters. Low-level wastes will be received in 55 gal. drums. Geotechnical factors taken into account by the design include: effects of heat on rock strength; rock movement; rock stability and clo[illegible] and groundwater movement. The rep[illegible]ory is assumed to operate in a retrievable mode for a period (currently planned at 5 years) to validate repository operations and safety. After this period, retrievability will be terminated and the storage rooms backfilled. Once all rooms are backfilled, surface facilities will be decommissioned and the shaft plugged and sealed. (Auth) (LKW)

Appendices describe hoisting systems, alternative canister emplacement systems, and job classification breakdown.

REPOSITORY; DESIGN; STORAGE, GEOLOGIC; SALT DEPOSITS; GEOLOGIC FORMATIONS; RETRIEVABILITY; THEORETICAL STUDIES; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, TRANSURANIC; CLADDING; SPENT FUELS; CANISTERS; WASTES, LOW-LEVEL;

<506>
Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Facility Construction Feasibility and Costs by Rock Type.

Y/OWI/TM-36/10; 120 pp. (1978, April)

This report presents the results of a study comparing the general engineering feasibility and unit costs associated with sinking shafts and mining storage rooms for a radioactive waste repository in salt, granite, shale, or basalt. The wastes considered are from three LWR fuel cycles: uranium and plutonium recycling, no recycling of spent fuel, and uranium-only recycling. The report includes a discussion of the general effects of rock characteristics on shaft and mine design, the application of these considerations to the specific designs developed for the draft GEIS, shaft and mine construction techniques, and the unit cost comparison. The repository designs upon which this comparison was based are presented in other volumes of this series. Results of this study show that when all five shafts for the facility are considered jointly, the cost of sinking the five shafts is $62,391/linear foot in salt, $32,391/linear foot in granite, $89,898/linear foot in shale, and $41,275/linear foot in basalt. Costs for shafts in salt, shale, and basalt are significantly higher than shaft costs in granite because shafts sunk in these three media are expected to traverse water-bearing zones which require extensive water control engineering during the shaft sinking operations. It should also be noted that the unit cost for shaft sinking in salt reflects the special water control techniques required to prevent dissolution of salt layers lying below water-bearing zones as the salt layers are penetrated. The effort required to control water inflows is expected to add approximately 45 weeks to the shaft construction schedule in salt, four weeks to the granite schedule, 33 weeks to the shale schedule, and 22 weeks to the basalt schedule. During the operation phase of a repository designed for Cycle II waste forms, total unit costs for mining were calculated to be $14.23/cu yd for salt, $23.77/cu yd for granite, $16.55/cu yd for shale, and $24.95/cu yd for basalt. These costs, which include mining, underground haulage, hoisting, surface disposal, power, equipment, and support, differ only minimally for repositories designed for Cycles I and III waste forms. (Auth) (LKW)

MINING; DRILLING; STORAGE, GEOLOGIC; REPOSITORY; SALT DEPOSITS; GRANITES; BASALTS; SHALES; IGNEOUS ROCKS; SEDIMENTARY ROCKS; ROCKS; DESIGN; ECONOMICS; TIME FACTOR; GROUND WATER; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; SPENT FUELS; WASTE STORAGE; WASTE DISPOSAL; SITE SELECTION; COST BENEFIT ANALYSIS

<587>
Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 10. Repository Preconceptual Design Studies: Granite.

Y/OWI/TM-36/10; 98 pp. (1978, April)

This report describes in detail a preconceptual design for a nuclear waste storage facility in granite. The facility design consists of several chambers excavated deep (2000 ft) within a geologic formation together with access shafts and supportive surface structures. Surface facilities and equipment are outlined, buildings are described, and detailed sequences of waste handling operations are given. Descriptions of facility power sources, electric services, utilization voltages, wiring, and sanitary waste disposal systems are provided. Surface facilities are designed to provide the following functions: receipt and unloading of wastes in specially designed containers from rail cars; decontamination of containers, if required; overpacking when needed; and preparation of waste for descent to the storage level. There are two basic waste handling streams: one for wastes requiring shielding and hot cell facilities (canistered waste) and one for wastes that do not (low-level waste). Five shafts are provided in the facility design: a canistered waste shaft to the storage level; a low-level waste shaft for transporting palletized drums of low-level waste to the storage level; a men and material shaft, also used for transporting excavated material to the surface; a ventilation supply shaft; and a ventilation exhaust shaft. Shaft boring and lining plans are described. All high-level wastes from IWR fuel cycles, PWR and BWR spent fuel assemblies, and intermediate-level TRU and cladding wastes will be received in 55 gal drums. Geotechnical factors taken into account by the design include: effects of heat on rock strength, rock movement, rock stability and closure, and groundwater movement. In granite, major losses of strength are expected to occur at around 450-500 C; thermal loading would be adjusted so that critical temperatures are not exceeded in large volumes of rock in the immediate vicinity of the waste canisters. Total groundwater inflows to the repository are expected to be from a few gal/min in the early years of operation to 50-100 gal/min final stages; these estimates do not take into account heat effects, and may be high. The repository is assumed to operate in a retrievable mode for a period (currently planned at 5 years) to validate repository operations and safety. After this period, retrievability will be terminated and the storage rooms backfilled. Once all rooms are backfilled, surface facilities are decommissioned, and the shafts plugged and sealed. (LKN)

Appendices provide a hoisting systems description and job classification breakdown summary.

GRANITES; ROCKS; IGNEOUS ROCKS; REPOSITORY; DESIGN; STORAGE, GEOLOGIC; THEORETICAL STUDIES; WASTES, RADIOACTIVE; WASTE STORAGE; WASTE DISPOSAL; METHODS; EQUIPMENT; GROUND WATER; HYDROLOGY; ROCK MECHANICS

<589>
Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 12. Repository Preconceptual Design Studies: Shale.

Y/OWI/TM-36/12; 95 pp. (1978, April)

This report describes in detail a preconceptual design for a nuclear waste storage facility in shale. The facility design consists of several chambers excavated deep (2000 ft) within a geologic formation together with access shafts and supportive surface structures. Surface facilities and equipment are outlined, buildings are described, and detailed sequences of waste handling operations are given. Descriptions of facility power sources, electric services, utilization voltages, wiring, and sanitary waste disposal systems are provided. Surface facilities are designed to provide the following functions: receipt and unloading of wastes in specially designed containers from rail cars; decontamination of containers, if required; overpacking when needed; and preparation of waste for descent to the storage level. There are two basic waste handling streams: one for wastes requiring shielding and hot cell facilities (canistered waste) and one for wastes that do not (low-level waste). Five shafts are provided in the facility design: a canistered waste shaft to the storage level; a low-level waste shaft for transporting palletized drums of low-level waste to the storage level; a men and material shaft, also used for transporting excavated material to the surface; a ventilation supply shaft; and a ventilation exhaust shaft. Shaft boring and lining plans are described. All high-level wastes from IWR fuel cycles, PWR and BWR spent fuel assemblies, and intermediate-level TRU and cladding wastes will be received in 55 gal drums. Geotechnical factors taken into account by the design include: effects of heat on rock strength, rock movement, rock stability and closure, and groundwater movement. In shale, major losses in strength are expected to occur at around 200-300 C; thermal loading would be adjusted so that critical temperatures are not exceeded in large volumes of rock in the immediate vicinity of the waste canisters. Depending on site-specific permeabilities, repository shafts will penetrate some waterbearing sandstone and gravel layers above the waste storage level. Inflow of water into unlined shafts would be tens to hundreds of gal/min, necessitating careful sealing of these aquifers. Estimated total groundwater inflows to the repository are from 50 gpm initially to 700-1500 gpm at the final stages of operation (not taking into account thermal effects). The repository is assumed to operate in a retrievable mode for a period (currently planned at 5 years) to validate repository operations and safety. After this period, retrievability will be terminated and the storage rooms backfilled. Once all rooms are backfilled, surface facilities are decommissioned, and the shafts plugged and sealed. (LKN)

Appendices provide a hoisting systems description and job classification breakdown summary.

<548>

REPOSITORY DESIGN AND ENGINEERING

<548> CONT.
SHALES; ROCKS; SEDIMENTARY ROCKS; REPOSITORY; DESIGN; STORAGE, GEOLOGIC; WASTE STORAGE; WASTE DISPOSAL; WASTES, RADIOACTIVE; THEORETICAL STUDIES; METHODS; EQUIPMENT; GROUND WATER; HYDROLOGY; ROCK MECHANICS

<549>
Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 18. Repository Preconceptual Design Studies: Basalt.

Y/OWI/TM-36/18; 98 pp. (1978, April)

This report describes in detail a preconceptual design for a nuclear waste storage facility in basalt. The facility design consists of several chambers excavated deep (2000 ft) within a geologic formation together with access shafts and supportive surface structures. Surface facilities and equipment are outlined, buildings are described, and detailed sequences of waste handling operations are given. Descriptions of facility power sources, electric services, utilization voltages, wiring, and sanitary waste disposal systems are provided. Surface facilities are designed to provide the following functions: receipt and unloading of wastes in specially designed containers from rail cars; decontamination of containers, if required; overpacking when needed; and preparation of waste for descent to the storage level. There are two basic waste handling streams: one for wastes requiring shielding and hot cell facilities (canistered waste and one for wastes that do not (low-level waste). Five shafts are provided in the facility design: a canistered waste shaft to the storage level; a low-level waste shaft for transporting palletized drums of low-level waste to the storage level; a men and material shaft, also used for transporting excavated material to the surface; a ventilation supply shaft; and a ventilation exhaust shaft. Shaft boring and lining plans are described. All high-level wastes from LWR fuel cycles, PWR and BWR spent fuel assemblies, and intermediate-level TRU and cladding wastes will be received in 55 gal drums. Geotechnical factors taken into account by the design include: effects of heat on rock strength, rock movement, rock stability and closure, and groundwater movement. In basalt, major losses of strength are expected to occur at around 450-500 C; thermal loading would be adjusted so that critical temperatures are not exceeded in large volumes of rock in the immediate vicinity of the waste canisters. Total groundwater inflows to the repository are expected to reach up to 265 gpm during later years of operation. The repository is assumed to operate in a retrievable mode for a period (currently planned at 5 years) to validate repository operations and safety. After this period, retrievability will be terminated and the storage rooms backfilled. Once all rooms are backfilled, surface facilities are decommissioned, and the shafts plugged and sealed. (LKM)

Appendices provide a hoisting systems description and job classification breakdown summary. Numerous tables of data provided.

BASALTS; ROCKS; IGNEOUS ROCKS; REPOSITORY; DESIGN; STORAGE, GEOLOGIC; WASTE STORAGE; WASTE DISPOSAL; WASTES, RADIOACTIVE; THEORETICAL STUDIES; METHODS; EQUIPMENT; GROUND WATER; HYDROLOGY; ROCK MECHANICS; CONTAINERS; MINES; MINING; NUCLEAR FACILITIES

<550>
Parsons, Brinckerhoff, Quade and Douglas, Inc., New York, NY.

Technical Support for Generic Environmental Impact Statement: Radioactive Waste Isolation in Geologic Formations. Volume 16. Repository Preconceptual Design Studies: Battelle Pacific Northwest Laboratory Waste Forms in Salt.

Y/OWI/TM-36/16; 153 pp. (1978, April)

This report describes in detail a preconceptual design for a nuclear waste storage facility in salt based upon Battelle Pacific Northwest Laboratory (BPNL) projections for the waste forms to be handled. The facility design consists of several chambers excavated deep within a geologic formation together with access shafts and supportive structures. Geotechnical factors taken into account by the design include: effects of heat on rock strength; rock movement; rock stability and closure; and groundwater movement. The repository is assumed to operate in a retrievable mode for a period (currently planned at 5 years) to validate repository operations and safety. After this period, retrievability will be terminated and the storage rooms backfilled. Once all rooms are backfilled, surface facilities will be decommissioned and the shafts plugged and sealed. Various modifications in this basic repository system would be required to handle the different waste package types projected by BPNL. In considering the alternative fuel cycles (Cycle I - total recycle (U and Pu), Cycle II - spent unreprocessed fuel, and Cycle III - U recycle only), cladding waste from Cycles I and III is packaged in 30 in. diameter x 10 ft long canisters in the BPNL study, and in 12 in. diameter x 10 ft long canisters in the Office of Waste Isolation (OWI) study; the BPNL study subdivides intermediate-level waste into 5 groups based on surface dose rate and packaging type. BPNL does not specify any low-level waste in Cycle II; BPNL canisters for PWR and BWR assemblies are smaller in diameter than those projected by OWI. BPNL data require the Canistered Waste Receiving Building to handle drums arriving in casks as well as canisters. BPNL waste transport projections require a reduction in railroad trackage serving the site and an increase in roadways and truck maneuvering and parking areas. The mine layout and operations would be modified because of increased space required for LLW and ILW from Cycles I and III, and because BPNL's packaging methodology results in HLW with an internal heat loading that increases with time, necessitating increased spacing for canisters in successive storage panels. Because of a shorter assumed aging period, BPNL Cycle II projections result in more spent fuel canisters received at the repository per year, reducing the operating life of the repository. (LKM)

REPOSITORY; DESIGN; STORAGE, GEOLOGIC; SALT DEPOSITS; GEOLOGIC FORMATIONS; RETRIEVABILITY; THEORETICAL STUDIES; WASTES, HIGH-LEVEL; WASTES,

<553> CONT.
INTERMEDIATE-LEVEL; WASTES, LOW-LEVEL; WASTES, TRANSURANIC; WASTES, RADIOACTIVE; CLADDING; SPENT FUELS; CANISTERS; WASTE MANAGEMENT; WASTE DISPOSAL; WASTE STORAGE; FUEL CYCLES; PACKAGING

<551>
Fittman, F.K., U.S. Atomic Energy Commission, Division of Waste Management and Transportation, Washington, DC.

Management of Commercial High-Level Radioactive Waste.

CONF-740346; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, R.E. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 1, (pp. 1-8), 166 pp.(1976)

Technology is presently available to allow the containment of high-level radioactive waste. Through proper waste management controls, the potential for public hazard and environmental damage can be reduced to acceptable levels. The AEC's program provides retrievable storage for solidified wastes in carefully maintained and monitored engineered facilities for several decades. During this period, investigation and demonstration of waste disposal in a deep, stable, geologic formation will continue. After the completion of an extensive testing and demonstration program establishing the acceptability of the method of permanent disposal, the radioactive waste will be removed from the retrievable surface storage facility and disposed of permanently. (Auth) (JHT)

WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE; WASTES, COMMERCIAL; CONTAINMENT; WASTE MANAGEMENT; WASTE DISPOSAL; RETRIEVABILITY; STORAGE, GEOLOGIC; DESIGN; BASINS, STRUCTURAL

<552>
Science Applications, Inc., Oak Ridge, TN.

Technical Support for Generic Environmental Impact Statement (GEIS): Radioactive Waste in Geologic Formations. Volume 1. Executive Summary.

T/OWI/TM-36/1; 60 pp.(1978, April)

This volume is the first of a 23-volume series on radioactive waste isolation in geologic formations. Preconceptual repository design studies have been performed in support of the Generic Environmental Impact Statement on Commercial Waste Management prepared under the auspices of the Department of Energy. The series provides a technical basis for the preconceptual designs, resource requirements, and environmental source terms associated with isolating commercial light-water reactor wastes in underground repositories in salt, granite, shale and basalt. Wastes are considered from three fuel cycles: uranium and plutonium recycling, no recycling of spent fuel, and uranium-only recycling. This volume briefly describes the Alternate Repository Preconceptual Design Studies that led to the generic repository designs and the determination of the repository's resource requirements and environmental effluents. (Auth) (RAF)

REPOSITORY; DESIGN; WASTES, SOLID; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, TRANSURANIC; SALT DEPOSITS; SALT DOMES; GRANITES; SHALES; BASALTS; WASTE DISPOSAL; SITE SELECTION; ROCKS; FUEL CYCLES; PLUTONIUM; URANIUM; REACTORS, LIGHT WATER; STORAGE, GEOLOGIC; SPENT FUELS; VITRIFICATION; EXCAVATION; GROUND WATER; LEACHING; RADIONUCLIDE MIGRATION; THERMAL PROPERTIES; CONTAINERS; RETRIEVABILITY; ROCK MECHANICS; ACCIDENTS; HAZARD ANALYSIS; HYDROLOGY; GEOLOGY; CONTAINMENT; EXPOSURE, OCCUPATIONAL; MONITORING; CRITICALITY; PIPES; RECOMMENDATIONS; WASTE STORAGE; WASTES, RADIOACTIVE

<553>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Description of Waste Isolation Facilities.

T/OWI/TM-44; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 4, (pp. 4-1 - 4-109), 837 pp.(1978, April)

This chapter describes design criteria for repositories, surface facilities, waste receiving shafts, and repository operations for each of the three fuel cycles in salt, argillaceous (shale), crystalline (granite), and volcanic (basalt) media. Ventilation systems, including subterranean and surface structures are described. Groundwater management and site preparation, including land improvements and utility systems necessary before operation, and environmental and industrial safety procedures necessary during operation (such as monitoring, waste treatment, confinement systems, and repository security measures), are discussed. (CAJ)

REPOSITORY; DESIGN; WASTE STORAGE; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; WASTE MANAGEMENT; SALT DEPOSITS; SHALES; GRANITES; BASALTS; SITE PREPARATION; SAFETY; MONITORING; WASTE TREATMENT; GROUND WATER; DRUMS; CONTAINERS; SHAFTS; MONITORING

<554>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Effluents from the Waste Isolation Facility.

T/OWI/TM-44; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 5, (pp. 5-1 - 5-52), 837 pp.(1978, April)

Waste effluents are divided into those associated with the construction and operation activities, and those concerned with the handling and storage of the waste itself. Radioactive releases during normal operations and isolation, accidental

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REPOSITORY DESIGN AND ENGINEERING

<554> CONT.
releases, and impacts and load limits of thermal effluents from waste decay heat are discussed. (CAJ)

References, flow charts, and tabular data are included.

WASTE STORAGE; WASTES, RADIOACTIVE; REPOSITORY; DESIGN; HAZARD ANALYSIS; STORAGE, GEOLOGIC; WASTE MANAGEMENT; EFFLUENTS; SAFETY; MONITORING; GRANITES; SHALES; BASALTS; WASTES, LOW-LEVEL

<555>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Environmental Monitoring.

Y/OWI/TM-44; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 7, (pp. 7-1 - 7-8), 437 pp.(1978, April)

This chapter contains a description of the general concepts that may apply to the proposed environmental monitoring program which will be established at each repository location. It includes radiological, meteorological, ecological, and geological-hydrological monitoring programs. (CAJ)

WASTE STORAGE; WASTES, RADIOACTIVE; REPOSITORY; DISPOSAL SITE; MONITORING; STORAGE, GEOLOGIC; WASTE MANAGEMENT; GEOLOGY; HYDROLOGY; METEOROLOGY; RADIOECOLOGY; RECOMMENDATIONS; ENVIRONMENT; METHODS; DESIGN; EFFLUENTS; RADIATION DETECTORS

<556>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Decommissioning.

Y/OWI/TM-44; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 8, (pp. 8-1 - 8-7), 437 pp.(1978, April)

This chapter describes some of the steps involved in decommissioning, which returns the repository surface area to as near to its original natural conditions and contours as is feasible, while maintaining monitoring for environmental control. Included are decommissioning of mine and surface facilities, (i.e., drainage and storage of contaminated liquids, decontamination and/or disposal of buildings, equipment, scrap materials, and mine residues), and site restoration. (CAJ)

MONITORING; DECOMMISSIONING; WASTES, RADIOACTIVE; REPOSITORY; WASTE MANAGEMENT; ENVIRONMENT; DECONTAMINATION; BURIAL; EQUIPMENT; WASTE DISPOSAL

<557>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Deep Geologic Alternative Actions.

Y/OWI/TM-44; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 9, (pp. 9-1 - 9-106), 437 pp.(1978, April)

The possibility of using subsurface facilities for interim retrievable storage (up to 25 years) of spent unreprocessed fuel (SNFF) assemblies is investigated. Two approaches are considered: reduced local thermal loading, and heat removal from the mine area through ventilation. Alternative designs to reduce acreage requirements, and preconceptual repository designs in salt for alternate waste forms are discussed in detail. Safety hazards such as accidental fires and thermal releases are considered. Alternatives for geologic disposal including deep continental formations and subsembed disposal are discussed. (CAJ)

Mine plans, graphs, and technical drawings of the underground storage facilities are presented. Tabular data are also included.

RECOMMENDATIONS; SITE SELECTION; WASTE STORAGE; WASTES, RADIOACTIVE; REPOSITORY; RETRIEVABILITY; STORAGE, GEOLOGIC; SPENT FUELS; EQUIPMENT; HAZARD ANALYSIS; DESIGN; WASTE MANAGEMENT; SALT DEPOSITS; GEOLOGIC FORMATIONS; TEMPERATURE; SAFETY; CONTAINERS; WASTES, LOW-LEVEL; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; SEA DISPOSAL

<559>
Science Applications, Inc., Oak Ridge, TN; Dames and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Potential Mechanisms of Containment Failure.

Y/OWI/TM-44; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 10 (pp. 10-1 - 10-21), 437 pp.(1978, April)

A summary of the results of a discussion in an earlier report on the potential causes of containment failure is presented. Manmade causes such as sabotage, warfare, and drilling, and natural causes such as meteorite damage, climate and other environmental factors, and geotectonic phenomena are discussed. (CAJ)

Tabular data and maps are included.

RECOMMENDATIONS; WASTE STORAGE; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; WASTE MANAGEMENT; HAZARD ANALYSIS; CONTAINMENT; CONTAINERS; CONTAINER INTEGRITY; ENVIRONMENT; GEOLOGY; ROCK MECHANICS; METEORITES; CLIMATE; EARTHQUAKES; ACCIDENTS

<558>
Science Applications, Inc., Oak Ridge, TN; James and Moore, White Plains, NY; Parsons Brinckerhoff Quade and Douglas, Inc., New York, NY.

Considerations Relevant to Provisional versus Final Storage.

Y/OWI/TS-44; Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 11, (pp. 11.1 - 11.9), 437 pp.(1978, April)

This chapter concerns the conditions under which retrievable and irretrievable isolation would be needed, and discusses the considerations of utilizing a deep geologic repository for extended (25-100 years) interim storage of spent fuel. (CAJ)

RECOMMENDATIONS; WASTE STORAGE; WASTES, RADIOACTIVE; REPOSITORY; DESIGN; STORAGE, GEOLOGIC; WASTE MANAGEMENT; SPENT FUELS; WASTE DISPOSAL; RETRIEVABILITY; COST BENEFIT ANALYSIS

<560>
Stottlemyre, J.A., G.W. Petrie, and W.F. Sullen, Battelle-Pacific Northwest Laboratories, Richland, WA.

Computer Enhanced "Release Scenario" Analysis for a Nuclear Waste Repository.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 521-531), 563 pp. (1979)

For long term safety of a repository such phenomena as long or short period geologic process, man-caused perturbations, and repository-induced events must be considered. This study presents an attempt at a lumped parameter simulation to quantify the isolation characteristics of a geologic system containing a repository. The computer program will estimate ranges of input data for the minimum time interval between backfill of a repository and a given type of accident, minimum path length from repository to biosphere, and maximum severity of a given type of breach. Two modes are used: parametric sensitivity analysis and "Monte Carlo" simulation. (WDV)

REPOSITORY; HAZARD ANALYSIS; COMPUTER CODES; SAFETY; GEOLOGY; BIOSPHERE; STATISTICS; MODELS, MATHEMATICAL

<561>
The Boeing Company, Boeing Engineering and Construction Division, Seattle, WA.

High-Level Waste Canister Emplacement and Retrieval Concepts Study.

Y/OWI/SUB-76/16509; 67 pp.(1976, September)

The primary task of this study was to develop storage concepts for the emplacement of high- and intermediate-level waste canisters in salt. Available assumptions were given on the canister, plug, sleeve, and overpack. In addition to the available assumptions, study ground rules were established through Boeing/Union Carbide Corporation coordination and agreement. Preliminary systems analysis work was done on the emplacement and retrieval of the canisters to gain insight into the total problem. System flow charts were developed to describe the basic canister handling operations required at the initial pilot plant. The initial systems planning revealed possible problems in emplacing and retrieving the quantity of canisters in the pilot plant during the given time frame. Canisters would have to be emplaced at a rate of 1.5/hr to achieve a satisfactory planned pilot plant operation. Four vertical, two horizontal, and a 45-degree concept were generated under this study. Only concepts with sleeves were developed as the salt may be structurally inadequate. The vertical concepts were found to be superior to the horizontal. The transporter handling system would be simpler, requiring only a hoist to attach to the canister lift pin for emplacement and retrieval. Horizontal concepts would require the transporter to provide full-length horizontal canister support after the canister has been rotated from the vertical position as well as a push/pull loading device to emplace and retrieve the canister. Vertical sleeve and plug concepts are recommended for follow-on development and analysis work. A sleeve with a plug that is mechanically inserted and withdrawn looks most promising at this time. The other vertical concepts have features which may be desirable in reducing the transporter time-line and shielding requirements. The following activities are recommended as follow-on to this study: 1) perform a thermal and radiation analysis of the canister from the time the shipping cask is opened above ground through loading on site, emplacement, and storage, 2) perform a time-line analysis of the handling and emplacement operation from the time a cask arrives at the repository to completion of emplacement, in order to provide bases for determining quantities of equipment, personnel, size of working areas, and design requirements to meet the planned disposal rate for the repository; and 3) look at the impact each of the seven storage concepts has on the total volume of mined space required for the repository. (LHH)

WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTE STORAGE; METHODS; DESIGN; SALT DEPOSITS; REPOSITORY; RETRIEVABILITY; CANISTERS; OVERPACKING; SLEEVES; THEORETICAL STUDIES; WASTE MANAGEMENT; WASTE DISPOSAL

<562>
Tierney, M.S., F. Lasso, and H.R. Shaw, Sandia Laboratories, Albuquerque, NM; U.S. Geological Survey, Menlo Park, CA.

An Analysis of the Geological Stability of a Hypothetical Radioactive Waste Repository in a Bedded Salt Formation.

CONF-781121, Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 501-511), 563 pp.(1979)

This paper is a report on the development of mathematical models used in preliminary studies of the long-term safety of

<552> CONT.
radioactive wastes deeply buried in bedded salt formations. Two analytical approaches to estimating the geological stability of a waste repository in bedded salt are described: a) use of probabilistic models to estimate the likelihoods of release of radionuclides from the repository through certain idealized natural and anthropogenic causes, and b) a numerical simulation of certain feedback effects of emplacement of waste materials upon ground-water access to the repository's host rocks. These models are applied to an idealized waste repository for the sake of illustration. The authors hesitate to draw conclusions about geologic disposal schemes on the basis of numerical results obtained with the present models for the idealized repository. In the absence of site-specific data general conclusions could be misleading. This work does allow one conclusion, however: probabilistic modeling and numerical simulation are useful tools for determining what are the most important scientific problems in the study of safety of geologically buried radioactive wastes. Furthermore, if probabilistic approaches to safety assessment are required for programmatic reasons, then the geological simulation method is an essential adjunct to the program and further development of simulation methods is encouraged. (Auth) (JCW)

WASTE STORAGE; WASTE DISPOSAL; STORAGE, GEOLOGIC; SALT DEPOSITS; MODELS; MODELS, MATHEMATICAL; WASTES, RADIOACTIVE; THEORETICAL STUDIES; REPOSITORY

<563>
Walsh, E.L., and E.A. Bathke, Human Sciences Corporation, Colorado Springs, CO.

Thermal Calculations in Shale.

Y/OWI/SUB-76/96502; K-76-1090(N); 36 pp. (1976, September 28)

Thermal calculations were performed for underground nuclear waste disposal sites assuming shale as the site material. Average thermal properties were used for zero inclined layered shale with waste disposal at a depth of 2000 ft below the surface. Thermal conductivity was assumed constant with temperature and time and no consideration was given for water loss. Vertical conductivity was assumed half of the horizontal conductivity. No allowance was made for formation of brick or ceramic in shale at elevated temperatures. Ambient temperature at a depth of 2000 ft was taken as 93 F. Coolant air was input at 60 F. Time dependent calculations were made for time dependent sources in the assumed geometry within the non-isotropic media. Thermal loads for times after emplacement of 1/2, 1, 2, 5, 10, and 30 years were calculated. Analysis of ground disposal temperatures resulting from emplacement of a total heat load of 960 kW using steady state calculations indicates: (1) no difference in thermal gradients for 2.25 or 5 kW containers beyond 1 or 2 source diameters, and (2) equilibrium temperature affected areas are at most 2500 ft horizontally and 2000 ft vertically from the source room. Fourier numbers related to the time to attain equilibrium indicate on the order of 1000 years to reach steady state at constant source strength. Thus, steady state temperatures are conservative by at least the source decay rate in the first century. Time dependent calculations over the first 30 years indicate: (1) maximum temperatures close to the canisters are reached approximately 2 years after the waste canisters are emplaced; (2) significant temperature increases are only within 100-300 ft. of the source room; (3) the governing criterion for personnel and equipment operation will be room floor temperature, which reaches 370 F maximum for the unventilated condition; (4) room air temperature can be reduced to reasonable levels with an approximately 2 day air conditioning cycle using 60 F air; (5) continuous room conditioning appears to be prohibitive for the assumed room design since source heat is sucked out of the ground around the room, overloading the conditioning unit. (Auth) (LKW)

THEORETICAL STUDIES; MODELS; MODELS, MATHEMATICAL; SHALES; REPOSITORY; STORAGE, GEOLOGIC; THERMAL PROPERTIES; CANISTERS; WASTES, RADIOACTIVE; DESIGN; WASTE DISPOSAL; WASTE STORAGE; COOLANTS

<564>
DeSteese, J.G., and R.E. Rhoads, Battelle-Pacific Northwest Laboratories, Richland, WA.

National Waste Terminal Storage Program: Potential Problems in the Waste Transportation System.

PNL-2444; 20 pp.(1977, December)

Potential problems are identified which may impact the planning, organization, and operation of nuclear waste transportation systems serving federal repositories. These system-level problems have the potential of seriously interfering with the overall NWI Transportation/Logistics Study objective of having a viable nuclear waste transportation system in 1985. This report includes recommended action and priority judgments to address these problems and minimize their impact. Potential problems associated with equipment and hardware are not covered. The potential problems identified as most important have consequences which may impact the overall state of future preparedness for transporting nuclear waste. Other important concerns relate to the imposition of unnecessarily severe and costly restrictions on nuclear waste transportation, public and carrier acceptance, and the involvement of interested parties in planning and decision-making. The major recommendation of this report is that the planning and development of the waste transportation system should be controlled by a central planning activity which anticipates the impact of uncertainties and undesirable events. (Auth)

WASTES, RADIOACTIVE; *REVIEWS; TRANSPORTATION; RISKS ANALYSIS; SAFETY; ACCIDENTS; CASES; STANDARDS, FEDERAL; WASTES, RADIOACTIVE; RECOMMENDATIONS; SOCIAL ASPECTS; THEORETICAL STUDIES

<565>
Fullwood, R.R., and A.J. Unione, Science Applications, Inc..

Transportation of Spent Fuel by Barge and By Rail: A Comparison of Risk.

CONF-790602; Proceedings of the 25th Annual Meeting of the American Nuclear Society, Atlanta, GA, June 3-8, 1979, (5 pp.).(1979, June 7)

This paper addresses the quantification of risk to the public for a projected 1985 reactor population from transporting spent fuel by an intermodal scheme - that is, via barges as the major mode of transport with railways being used between the reactor and/or reprocessing site and a navigable waterway. The intermodal system considered uses presently available barge transportation and casks designed for rail car transport. Risk calculations are made for two generic types of railcar-sized casks currently licensed by the Nuclear Regulatory Commission (NRC). The "dry" cask considered is the NLI 10/24, while the "wet" cask is the IF 300. Three categories of air release were defined for the "dry" cask: (1) minor air release -- loss of containment integrity, but no rod failures beyond those assumed to occur during reactor operation (less than 1%); (2) severe air release -- loss of containment integrity with significant mechanical rupture (about 50%) of fuel cladding; (3) very severe air release -- loss of containment integrity, significant mechanical failure of fuel claddings, and post-impact overheating of the failed rods in a sustained thermal environment. A fourth air release category was postulated for the "wet" cask, as well as a water release category for both casks. Risk estimates were based on frequencies of each accident release event determined from fault trees, the frequency per reactor year of each accidental release event, and the resultant risk to the public per reactor year using estimates of the consequences of each release event in man-rem. Methodology for determining air release consequences was taken from WASH-1400, while methodology for determining water release consequences was taken from NRC regulatory guides. A similar risk estimate was compiled for spent fuel handled exclusively by rail. Results showed that average risk to the public from intermodal transport is 1.2 X 10(E-6) man-rem/R-yr from all postulated release accidents, and 6.0 X 10(E-7) man-rem/R-yr for rail transport alone. The three-fold difference is due entirely to the added risk of release to water from barge transport, and is not considered statistically significant since the uncertainty in results is approximately one order of magnitude. Estimated risk to the U.S. public from a spent fuel transportation accident for a 1985 reactor population is 7.9 X 10(E-5)-1.3 X 10(E-4) man-rem/yr; the high estimate corresponds to servicing 40% of reactor sites in 1985 with intermodal transport, while the lower estimate corresponds to a 5% intermodal servicing. Comparison of these results with those of WASH-1400 indicates that the risk of spent fuel transportation by any mode is smaller than the risk from light water reactor operation by a factor of more than 10(E+7). (Auth)(LKM)

See also SAI-054-77-PA, "A Generic Assessment of Barge Transportation of Spent Nuclear Fuel," A.J. Unione, A.A. Garcia, and R. Stuart, June 1978.

TRANSPORTATION, BARGE; TRANSPORTATION, RAIL; HAZARD ANALYSIS; ACCIDENTS; CASES; SPENT FUELS; RADIATION HAZARDS; POPULATIONS; THEORETICAL STUDIES; RADIATION DOSE; DOSE RATE; DOSE COMMITMENTS

<566>
Joy, D.S., and B.D. Holcomb, Oak Ridge National Laboratory, Chemical Technology Division, Pilot Plant Section, Oak Ridge, TN.

Logistics Models for the Transportation of Radioactive Waste and Spent Fuel.

ORNL/TM-6192; 89 pp.(1978, March)

Logistics models for the transportation of low-level waste and spent fuel have been developed at ORNL in cooperation with the Office of Waste Isolation. The Low-Level Waste Model evaluates optimal transportation policies for shipping waste directly from the sources to a final destination without intermediate stops. The Spent Fuel Logistics Model evaluates optimal transportation policies for shipping unreprocessed spent fuel from nuclear power plants (1) indirectly, to an Away-From-Reactor (AFR) storage facility, with subsequent transshipment to a repository, or (2) directly to a repository. The models permit

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TRANSPORTATION TECHNOLOGY

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determination of shipping schedules, costs, transportation modes, minimal transportation equipment, minimal personnel exposure, optimal routes, etc., allowing repositories to balance work loads, evaluate surge capacity requirements, and estimate projected shipping cask fleets. The programs are tailored to utilize information on the types of wastes being received, location of repositories and waste generating facilities, shipping distances, time required for a given shipment, availability of equipment, above-ground storage capabilities and locations, projected waste through-put rates, etc. While each model was designed to simulate a particular waste movement, either model could be used to optimize the shipment of any commodity which would be shipped within the constraints outlined. In addition to these models, a Nuclear Reactor Data Base describing all U.S. nuclear power plants operating, under construction, or on order as of March 1, 1977 has been developed, together with a Spent Fuel Shipment Program to predict annual spent fuel shipments from U.S. reactors. (Auth) (LXH)

WASTES, LOW-LEVEL; FUELS; TRANSPORTATION; MODELS; WASTE STORAGE; WASTE DISPOSAL; WASTE MANAGEMENT; COST BENEFIT ANALYSIS; WASTE VOLUME; SPENT FUELS; MODELS, MATHEMATICAL

<567>

Vasilyanovskii, B.T., and A.I. Vamin.

Experience in the Organization of Sanitary Radiometric Control at the Stage of Transporting Solid Radioactive Wastes.

CONF/Pr-4522; 8 pp.; Gigiena i Sanitariya 41:96-97, (1976)

Methods for monitoring the air around the transportation vehicle and smear samples are presented. High-capacity aspiration units are used to take samples of the air in the vehicle cabins during transportation of radwaste as well as loading and unloading, at various distances from the loaded or empty vehicle, and at the moment the dust cloud is formed by the moving truck. A special device was designed to take smear samples with a constant force of 3 kg/sq cm. The information obtained from the use of these methods made it possible to identify the role of solid wastes in the spread of radioactive substances beyond the confines of enterprises and burial sites as well as evaluate the health and radiation situation in waste transportation. (MOW)

ATMOSPHERE; CONTAMINATION; FIELD STUDIES; MONITORING; TRANSPORTATION; HAZARD ANALYSIS; WASTES, RADIOACTIVE; INSTRUMENTS; EQUIPMENT; METHODS

<569>

Hess, H., and S.C. Beckman, Sandia Laboratories, Albuquerque, NM.

Fire Protection Device for LMFBR Spent Fuel Shipping Casks.

SAND-77-0723; 18 pp.(1977, August)

This report describes the analysis and experimental evaluation of a device to provide fire protection for a shipping cask used to transport liquid metal fast breeder reactor (LMFBR) fuel rods. Thermal analyses of various fire protection schemes were conducted by means of the finite difference code, CINDA. The choice of materials for the cask body was determined to be less important than the dimensions and number of cooling fins attached to the cask. Of several protection methods considered, radiation/convection shields between the cooling fins looked most attractive; these were tested on a quarter-scale cask model. The shields consisted of bimetallic bands which expanded when heated, blocking the fire, and retracted when cooled. This automatic reversibility would allow the fins to dissipate internally generated heat of a full-size cask once the fire was out. Tests showed that, even in an asymmetric fire exposure, the bands expanded to provide protection to the cask. The directly exposed surface of the cask model, protected this way, reached a temperature of only 180 degrees C compared with 295 degrees C in the unprotected state, when subjected to a butane flame for 0.5 h at a distance of 15 cm. Greater relative effectiveness could be expected in an engulfing fire in which the bands would expand more symmetrically. (Auth)

Temperature

CASKS; FIRE HAZARDS; SAFETY; FUEL RODS; COMPUTER CODES; REACTORS, LIQUID METAL FAST BREEDER; TRANSPORTATION; ACCIDENTS; THERMAL PROPERTIES; HEAT RESISTANCE; HEAT EXCHANGE; MODELS; THERMAL EXPANSION; THEORETICAL STUDIES; LABORATORY STUDIES; TEMPERATURE

<569>
Aaron, W.S., T.C. Quinby, and E.H. Kobisk, Oak Ridge National Laboratory, Oak Ridge, TN.

Cermets for High Level Waste Containment.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 191-193), 563 pp. (1979)

Results of a study on cermet as a method of fixing high-level waste are reported. This waste form consists of a continuous iron-nickel base metal matrix containing small particles of fission product oxides. Coprecipitation of waste and specific additives from a molten urea solution produces this waste form, which provides an intimate, homogenous mixture of all components calcined to their respective oxides. Metal species such as iron and nickel can be then reduced to metal from their oxides, and the oxide-metal mixture leads itself to be densified. Densification is accomplished by hot pressing or extruding with a binder, and then sintering. Several advantages result from this method: thermal conductivities in the range of 0.02 to 0.04 cal/cm sec degree C and waste loading levels up to 75 wt% waste and 15 wt% contaminated metals and only 20 wt% noncontaminated additives. Due to these properties much higher storage volume reductions are possible. Preliminary leach tests indicate the cermets possess good resistance. (SDW)

Thermal Conductivity

BINDERS; DENSIFICATION; ENCAPSULATION; FISSION PRODUCTS; FIXATION; IRON; NICKEL; ALLOYS; PRECIPITATION, CHEMICAL; THERMAL PROPERTIES; VOLUME REDUCTION; WASTE TREATMENT; METALS; CERMET; LEACHING; WASTES, HIGH-LEVEL; METHODS; LABORATORY STUDIES

<570>
Alexander, B.M., D.H. Batchelder, J.W. Doty, and E.L. Lewis, Mound Facility, Miamisburg, OH.

Volume Reduction System for Solid and Liquid Transuranic Waste from the Nuclear Fuel Cycle: January-March 1978.

MLM-2517; 16 pp. (1978, May 5)

The laboratory-scale cyclone incinerator at the Mound Facility began operation in January 1978 to study the effects of burning waste contaminated with radionuclides other than Pu 239, primarily fission products. The first two months of operation revealed several areas for improvement. Experiments were conducted on pressed ash/cement and sludge/cement pellets to evaluate their potential effectiveness for TRU waste immobilization. Tests of compressive strength, mechanical stability, density effects of wet versus dry curing, and other physical and chemical aspects of the ash/cement matrix were carried out. Leachability and compressive strengths of pressed sludge/cement matrices were also tested. Waste reduction efficiency of ash/cement pelletization was approximately 95% in weight reduction and 97% in volume reduction. Average compressive strengths ranged from as low as 2880 psi for dry cured 60% and 80% ash matrices to as high as 8580 psi for wet cured 100% Portland PA cement. Wet cured pellets consistently exhibited greater compressive strength. Density of pellets increased as ash content decreased, densities ranging from 93 to 99% of the 0% ash cement matrix. Leach rate for a 20% sludge mixture was 3.55 X 10(E-6) g/sq cm/hr; for 15% sludge the rate was 5.25 X 10(E-6) g/sq cm/hr. Elemental compositions and particle sizes of ash types, water/cement ratios, and mechanical weight losses are given. (LEN)

Leaching Rate; Density

INCINERATION; CEMENTS; ASHES; DESIGN; EQUIPMENT; LABORATORY STUDIES; PARTICLE SIZE; PHYSICAL PROPERTIES; SLUDGES; TRANSURANICS; WASTE TREATMENT; WASTES, RADIOACTIVE; WASTES, SOLID; WASTES, TRANSURANIC; IMMOBILIZATION; LEACHING; VOLUME REDUCTION

<571>
Alexander, B.M., E.L. Lewis, J.W. Doty, and D.H. Batchelder, Mound Facility, Miamisburg, OH.

Volume Reduction System for Solid and Liquid Transuranic Waste from the Nuclear Fuel Cycle: July-September 1978.

MLM-2539; 17 pp. (1979, November 16)

The laboratory cyclone incinerator has been developed to study the effects of burning wastes containing radionuclides other than plutonium 238. These isotopes must be efficiently removed from the off-gas if combustible waste containing fission products is to be incinerated. Modifications to the original incinerator design were made to increase combustion efficiency, to improve particulate and gaseous pollutant removal in the off-gas system, to reduce energy consumption, and to decrease retention of material in the process system. Further changes contemplated will permit the safe burning of feed material spiked with radioisotopes. During the quarter the matrix for bone char and cement mixtures was investigated. Limited work was also done on determining the best matrix for sludge-cement pellets. The new glovebox line for immobilizing ash produced from incineration of <10 nCi Pu 238/g waste was completed. Extensive work was done to develop an acceptable sintering procedure. Almost 200 pellets were manufactured, and sample pellets were tested for crush strength and thermal stability. (Auth)

VOLUME REDUCTION; PLUTONIUM; WASTES, RADIOACTIVE; CEMENTS; INCINERATION; BONE CHAR; WASTES, TRANSURANIC; LABORATORY STUDIES

<572>
Ames, L.L., Jr.

Mineral Reaction Work at Hanford.

TID-7600; Use of Inorganic Exchange Materials for Radioactive Waste Treatment, D.N. Jamison, et al. (Eds.), Proceedings of a Working Meeting, Washington, DC, August 13-16, 1962. U.S. Atomic Energy Commission, Division of Technical Information, Washington, DC, (pp. 115-150), 239 pp. (TID-7600).(1967, January)

<572>

WASTE PRODUCTION

<572> CONT.

Results of studies on mineral reactions in an effort to remove specified radioisotopes from radwastes are presented. Only reactions involved with ion exchange are considered. Used in the studies were Linde AW800 (synthetic erionite), AW500 (synthetic chabazite), norton zeolon and Linde AW300 (synthetic mordenite), Linde 4A, and Linde 13X (synthetic faujasite). The total exchange capacity for each of these is Linde 4A, 3.9; 13X, 3.9; Linde AW 500, 2.2; Linde AW 800, 2.0; zeolon, 1.9; and Linde AW 300, 1.6 meq/g. For Linde 13X the mass action coefficient is about 12. This coefficient is defined as the fraction of Sr in the zeolite over the fraction of Na or the zeolite squared. A sample calculation of the mass action coefficient is included. Equilibrium curves indicated the total capacities listed are close to those measured in the laboratory. The two types of particle diffusion were indicated during kinetics studies. These two types of diffusion control the loading rates. The upper limiting case is particle diffusion where the diffusion within the particle itself becomes rate determining. The lower limiting case is diffusion across a solution film on the outside of the particle. This situation occurs in the region where high cation concentrations are present. Examples as well as the mathematics of the diffusion are given. (KDV)

A large number of graphs and charts are used to present pertinent data and information.

Mass Action Coefficient; Ion Exchange Capacity

SORPTION; BREAKTHROUGH DISTRIBUTION; CLINOPTILOLITE; DECONTAMINATION; EQUILIBRIUM CONSTANT; ION EXCHANGE CAPACITY; LABORATORY STUDIES; MASS BALANCE; WASTE TREATMENT; ZEOLITES; WASTES, RADIOACTIVE; WASTES, LIQUID; SOLIDIFICATION; ENCAPSULATION

<573>

Azamaeva, L.G., A.N. Kondrat'ev, I.I. Kryukov, V.V. Kulichenko, and Y.P. Martynov, V.G. Khlopin Radium Institute, Leningrad, USSR; All-Union Scientific Research Institute of Inorganic Materials, Moscow, USSR.

Above-Ground Storage Facilities for Vitrified Highly Active Wastes

BNWL-tr-189; 22 pp.(1976, July 23)

Two types of storage facilities for highly active vitrified wastes are examined in the present report: those with air cooling and those with water cooling of the solidified wastes. The results of thermophysical calculations are presented for vitrified wastes with various physical properties. The designs of storage facilities for vitrified wastes from a reprocessing plant and an economic comparison of the alternatives considered are presented. Simultaneously with the development of methods of solidifying radioactive liquid effluent, methods for the reliable and safe storage and burial of solidified wastes are being elaborated in the Soviet Union and other countries. The problem of the heat of radioactive decay occupies a special place in the selection of storage conditions for solidified wastes formed as a result of the vitrification of highly active liquid effluent from fuel-element reprocessing plants. In order to ensure that the temperatures of the solidifed wastes do not exceed the permissible levels for the given material, the decisive dimensions of tanks containing wastes or the specific heat release of the wastes must be limited. The present report illustrates the influence of specific heat release and permissible storage temperatures of vitrified wastes and of the methods used to cool the wastes on the required degree of curtailment of the decisive dimensions of waste-containing tanks. The influence of the degree of concentration of liquid effluent in the process of waste vitrification on the economics of their storage in various types of facilities also is considered. The calculations were made for vitrified wastes from a reprocessing plant with a capacity of 5 tons of uranium per day. Fuel elements from a water-moderated water-cooled power reactor are sent for reprocessing after a holding period of the order of 2(E+x) years. The volume of the facility is designed to be filled over a period of 5 years. It is accepted that the degree of concentration of liquid effluent during vitrification, and consequently the specific heat release of the glass-like products entering the storage facility, vary over a broad range. The values of the specific heat release of solidified products lie in the range 5 x 10(E+3) to 10(E+5) w/cu m. The volume of the stored product varies in inverse proportion to the specific heat release from 9400 cu m to 870 cu m. The total heat release in a completely filled storage facility for vitrified wastes is 20 Mw. (Auth)

WASTES, SOLID; WASTES, HIGH-LEVEL; GLASS; WASTE STORAGE; WASTES, LIQUID; REPROCESSING; FUEL ELEMENTS; TANKS; ECONOMICS; COOLANTS; CONTAINERS; STEELS; FILTERS; HEAT EXCHANGE; HYDROGEN; FABRICATION; SOLIDIFICATION; RADIOACTIVE DECAY; HEAT SOURCES; VITRIFICATION; STORAGE, ABOVEGROUND

<574>

Bahr, W., W. Hild, S. Drobnik, L. Kahl, H. Kelm, W. Kluger, H. Krause, W. Finsterwalder, and W. Huth, Gesellschaft fur Kernforschung mbH, Karlsruhe, German Federal Republic; Gesellschaft zur Wiederaufarbeitung von Kernbrennstoffen mbH, Karlsruhe, German Federal Republic; Nukem, GmbH, Wolfgang, German Federal Republic.

Recent Experiments on the Treatment of Medium Level Wastes and Spent Solvent and on Fixation into Bitumen.

CONF-760310; STI/PUB/433; IAEA-SM-207-81; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 2, (pp. 133-142), 824 pp.(1976)

The Nuclear Research Center at Karlsruhe has been conducting research and development work on radioactive waste volume reduction and fixation via bitumenization since 1972. Tests of the radiation stability of bitumenized wastes up to integrated doses on the order of 10(E+8) rads indicate that radiolytic H2 production is in linear proportion to the total dose, averaging 0.4-0.5 ml H2/g product per 100 Mrad; radiation-induced porosity poses no problem. Leaching of products with a 5 mm coating of pure bitumen occurred at a rate of about 3 x 10(E-8) g cm(E-2) d(E-1) after 5 years, versus

<574> CONT.
10(E-4)-10(E-5) g cm(E-2)d(E-1) without the coating. Lab- and pilot-scale work on the denitration of medium-level waste streams and evaporator concentrates by reaction with formic acid is described. Since part of the medium-level waste to be processed could be above the acceptable activity concentration for safe bituminization, research is being done on the separation of fission products, corrosion products, and actinides from the waste by precipitation and adsorption processes, leaving the remaining solution with an activity of about 0.01 Ci/l and most of the soluble salts. Work to date has produced decontamination factors on the order of 10(E+2)-10(E+3). Recycling of kerosene and TBP from organic waste solutions is discussed; more than 100 cm a of waste have been reprocessed batchwise, and continuous operations are planned. (LKN)

Porosity; Leaching Rate

VOLUME REDUCTION; BITUMINIZATION; LABORATORY STUDIES; WASTES, INTERMEDIATE-LEVEL; EVAPORATOR CONCENTRATE; FISSION PRODUCTS; ACTINIDES; PRECIPITATION, CHEMICAL; ADSORPTION; SALTS; DECONTAMINATION FACTORS; ORGANIC COMPOUNDS; FIXATION

<575>
Barney, G.S., Atlantic Richfield Hanford Company, Chemical Technology Laboratory, Richland, WA.

Fixation of Radioactive Waste by Hydrothermal Reactions with Clays.

CONF-740386; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, N.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 8, (pp. 108-125), 166 pp. (1976)

A low-temperature process for conversion of radioactive sodium salt wastes into solid, relatively insoluble, thermally stable sodium aluminosilicates is described. The reaction of the waste in aqueous solution with powdered clays, such as kaolin, bentonite, halloysite, or dickite produces small crystals (0.5 microsoters) of cancrinite and traps the radioactive salts in a crystal lattice of the cancrinite. The approximate chemical formula of the cancrinite produced is 2(NaAlSiO4) x salt, yH2O, with x=0.52 and y=0.68 when the entrapped salt is NaNO3. The stoichiometry requires two moles of NaOH for each mole of cancrinite formed. Reaction occurs at temperatures as low as 30 degrees C, but there is a large temperature effect on the rate. The half-times for the reaction of kaolin with standard waste at 100, 75, and 50 degrees C, respectively, are 1, 10, and 150 hr. The corresponding half-times for the bentonite reaction are 1, 9, and 60 hr. The cancrinite product has a low leach rate with values obtained by leaching with distilled water in the range of 10(E-7) to 10(E-9) g of sample/square cm-day. Diffusion of cesium ion from the product into the leachant is slower than for the sodium ion. The melting range of cancrinite products is about 900-1200 degrees C. Zeolitic water is removed at 100-300 degrees C and at about 700 degrees C the trapped NaNC2 and NaNO3 begin to decompose to Na2 and nitrogen oxides. A change in crystal structure to nepheline occurs at 530-595 degrees C. The thermal conductivity of cancrinite products ranges from 0.11 to 0.67 Btu/(hr)(square ft)(degrees F/ft). These values are thought to be high enough to prevent large temperature gradients in products made for Hanford stored waste. The mechanical strength of the product can be improved by adding hardeners and binders. Various inorganic cements, including Portland cement can be used to bind the cancrinite crystals to make a stronger product. Organic polymers can also be used. The volumes of the clay and waste solution are nearly additive. Fired clays appear to give the smallest volume increase. (JRT)

SALTS; WASTES, RADIOACTIVE, WASTES, HIGH-LEVEL; CANCRINITE; BENTONITES; LEACHING; SOLIDIFICATION; KAOLIN; HALLOYSITE; DICKITE; CLAYS; CEMENTS; BINDERS; THERMAL CONDUCTIVITY; CESIUM; CHEMICAL ANALYSIS; LEACHING; THERMAL PROPERTIES

<576>
Beeton, W.A.

Adsorption Studies by the U.S. Geological Survey.

TID-7684; Use of Inorganic Exchange Materials for Radioactive Waste Treatment, D.K. Jamison, et al. (Eds.), Proceedings of a Working Meeting, Washington, DC, August 13-14, 1962. U.S. Atomic Energy Commission, Division of Technical Information, Washington, DC, (pp. 211-222), 238 pp. (TID 7684)-(1963, January)

An experiment designed to test the adsorption of dissolved ions by a number of minerals by varying the calcium and sodium concentration of the solutions was run. Large increases of the sodium concentration resulted in a reduction of the amounts of calcium and strontium adsorbed on the minerals with a corresponding reduction of the distribution coefficient for strontium. Using the double-layer and mass action theories two equations are derived to explain the ion exchange. The two equations indicate that the fraction of sodium saturation depends on the surface charge density of the mineral instead of the cation exchange capacity. Montmorillonite and kaolinite were the two adsorbents. It was found that sodium has only a small effect on the distribution coefficient of strontium. An attempt was also made to correlate distribution coefficient values with exchange capacity of a material. (NDV)

Graphs illustrating the distribution coefficients with respect to ion concentration are presented.

Distribution Coefficient; Ion Exchange Capacity

ADSORPTION; CALCIUM; CATION EXCHANGE CAPACITY; DISTRIBUTION COEFFICIENT; EQUATIONS; IONS; LABORATORY STUDIES; MASS BALANCE; MONTMORILLONITE; KAOLINITE; STRONTIUM; CALCIUM; THEORETICAL STUDIES; SOLIDIFICATION; ENCAPSULATION

<577>
Bibler, N.E. Savannah River Laboratory, Aiken, SC.

<577>

WASTE PRODUCTION

<577> CONT.
Radiolytic Gas Production during Long-Term Storage of Nuclear Wastes.

DP-MS-76-41; 15 pp.(1976)

Gas evolution by in situ radiolysis of sealed solidified nuclear wastes during long-term storage, which could conceivably breach containment, is being studied at Savannah River Laboratory. Candidate waste forms, including cement containing simulated fission product sludges, vermiculite containing organic liquids, and cellulosics contaminated with alpha-emitting transuranic isotopes, have been irradiated with Co 60 gamma and Cm 244 alpha radiation. For cement waste forms exposed to gamma radiolysis, an equilibrium H2 pressure was reached that was dose rate dependent. Extrapolation to the calculated dose rate for actual waste indicates that for irradiation times of a few hundred years, during which radiolysis is due primarily to decay of Sr 90 and Cs 137, H2 equilibrium pressure will be 8-28 psi. For alpha radiolysis, equilibrium was not reached. With organic wastes (n-octane on vermiculite), H2 and traces of CO2 and CH4 were produced, and O2 was consumed with both radiations. The ratio of H2/CO2/CH4 was nominally 1.0/0.03/0.01. Only energy absorbed by the organic material was effective in producing H2. At low dose rates with both radiations G(H2) was 8.5 and G(-O2) was 5.0. H2 pressure did not attain equilibrium up to 200 psi, the highest pressure tested. This implies that in sealed containers of this waste type, very high H2 pressures could be attained. The extent of this pressurization is calculated to be at least 200 psi during long-term storage. For cellulosic material, H2, CO2, and CO were produced in the ratio 1.0/0.7/0.3, and O2 was consumed. With alpha radiolysis, G(gas) was dose dependent; measured values ranged from 2.2 to 0.6 as the dose increased. The amount of H2 produced in a waste container is currently being held to a minimum by limiting the amount of radioactivity stored in that container. H2 generation can be eliminated completely by storing the waste in the absence of water or organic material; the method being evaluated is to calcine the fission product sludges and incorporate them in glass. For organic waste, a method being evaluated is to incinerate the material and incorporate the ash in glass. (Auth) (LKM)

RADIOLYSIS; WASTES, SOLID; WASTES, RADIOACTIVE; CEMENTS; SLUDGES; VERMICULITE; ORGANIC COMPOUNDS; WASTES, LIQUID; TRANSURANICS; ALPHA PARTICLES; RADIATION, GAMMA; GASES; PRESSURE EFFECTS; HYDROGEN; CALCINATION; VITRIFICATION; INCINERATION; ASHES; LABORATORY STUDIES; FISSION PRODUCTS

<578>
Blane, D.E., and W.R. Herald, Mound Facility, Miamisburg, OH.

Removal of Plutonium from High-Level Caustic Waste Solutions Using Bone Char - Pilot Study.

MLM-2530; 10 pp.(1978, October 2)

Pilot studies have been completed using 275 micro-meter bone char as an absorbent for the removal of plutonium from Mound Facility caustic waste solutions. Bone char is the commercial name for a natural product made from granulated cattle bone and is a form of calcium hydroxyapatite [Ca10(PO4)6(OH)2]. The waste solutions emanate primarily from the three following sources: (1) a 4% NaOH corrosive vapor scrubber solution from the corrosive vapor system constituting about 90% of the volume; (2) filtrates from the NH4OH precipitation of plutonium constituting about 8% of the volume, and (3) scrubber solutions for off-gases from an incinerator, analytical solutions, and other miscellaneous caustic solutions, constituting about 2% of the total volume. These sources generated about 1600 gallons of caustic waste during the calendar year 1977. The waste stream was characterized by radioactivity levels in the range of 0.5 to 1.3 mg of plutonium/liter (maximum near 8 mg/liter), a pH of 8 to 13, and a moderate amount of suspended solids. During the experiment, the waste solutions were adjusted to an approximate pH of 8 before they were passed through the columns of bone char. Overall decontamination factors of about 5 x 10(E+9) were obtained in the study and a tentative design of a caustic waste treatment system utilizing cartridge filters and bone char columns was made. This proposed system has economic advantages over the current disposal methods and it offers a 10:1 reduction in the volume of solid waste that must be buried. (JHT)

PLUTONIUM; ABSORPTION; BONE CHAR; LABORATORY STUDIES; VOLUME REDUCTION; WASTE TREATMENT; pH; WASTES, RADIOACTIVE; DECONTAMINATION; COST BENEFIT ANALYSIS

<579>
Blomeke, J.O., and W.D. Bond, Oak Ridge National Laboratory, Oak Ridge, TN.

High-Level Waste Management Research and Development Program at Oak Ridge National Laboratory.

CONF-740386; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, W.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 6, (pp. 84-92), 166 pp.(1976)

Projections of wastes to be generated through the year 2000 present a problem of impressive size and complexity. It is estimated that the metric tons of spent fuels to be reprocessed annually will increase from 1700 in 1980 to 22,000 in the year 2000. Generation of concentrated high-level liquid waste will increase from 170,000 gal/year to 2.2 million gal/year in 2000. It is assumed that the actinide elements in the waste consist of 3.5% of the uranium and plutonium originally in the spent fuels plus all other heavy elements formed by transmutation of the uranium and plutonium in the reactors. Investigations of the technical feasibility of removing actinide elements from wastes to render the residuals more manageable in terms of hazards and storage requirements indicate that they can be removed from wastes by the minimally desired factors of 10(E+2) to 10(E+4). Natural salt formations are believed to offer the best prospects for disposal of high-level wastes, although other geologic formations, such as shales, granite, and limestone are also being studied. (Auth) (JHT)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, LIQUID; FUEL ELEMENTS; ACTINIDES; PREDICTIONS; WASTE STORAGE; SALT DEPOSITS; SHALES;

<579> COST.
SEDIMENTARY ROCKS; IGNEOUS ROCKS; GRANITES; LIMESTONES; WASTE MANAGEMENT; PARTITIONING; STORAGE, GEOLOGIC

<580>
Eckmer, W.F., and W.A. Ross, Battelle-Pacific Northwest Laboratories, Richland, WA.

Current High-Level Waste Solidification Technology.

BNWL-SA-5972; CONF-761038; Energy, Proceedings of the Third Annual DOE-NRC Conference, Rolla, MO, October 12-14, 1976, (9 p.). (1976)

Many nations are preparing plans and processes for solidification and disposal of high-level waste. Currently well developed processes are described and discussed: fluidized bed calcination, with and without vitrification; spray calcination with in-can melting; continuous ceramic melter; and the French, German, and British vitrification processes. The fluidized bed calciner converts liquid waste to a mixture of powdery solids and granules in the size range of 0.05-0.5 mm, with calcine density from 2.0-2.8 g/cm cu, and the specific volume 80/MTU. This process is reasonably flexible as to types of waste it can calcine. A collapsed bed is considered by many to be the maximum credible accident, which can result in a molten core surrounded by sintered calcine. However, the condition is preventable and correctable. When the calcine produced is transferred directly into a post-treatment step such as vitrification, nonconventional calciner operating techniques can be used. One such technique is termed the continuous inert bed process, wherein operability is simplified because bed particle size control is not required; also a low inventory of fission products can be maintained. The calciner has been successfully coupled to in-can or continuous ceramic melters to produce borosilicate glass. The spray calciner coupled to the in-can melter is the mainline process at Battelle. This system has proven simple and reliable for vitrification of a wide variety of nuclear wastes, even those in which over 90% of the cations are sodium. Liquid waste is sprayed into a hot, 700 degree C chamber where it is calcined and the calcine dropped directly into the in-can melting canister. The fine particles (mean diameter 10 um) react rapidly with the glass formers in the melter. In 1973, Battelle began developing a joule-heated, ceramic-lined melter, which typically exhibits low off-gas effluents, high capacity, long life, and high glass quality. It is simply a ceramic-lined, glass-filled cavity in which electric energy is dissipated in the molten glass between immersed electrodes. A pot vitrification process termed PIVER, and another vitrification process incorporating a rotary kiln calciner and continuous metallic melter were developed in France. One rotary kiln unit can calcine 30 to 40 l/hr of concentrated liquid high-level waste; thus 2 or 3 units could handle the wastes generated by a 5 ton/day nuclear fuel reprocessing plant. In Germany, the vitrification process termed VERA is the farthest advanced of the processes being developed there. The process involves 3 steps: liquid waste denitration, spray calcination, and melting to form a glass casting. An alternative glass packaging technique called PAKELA is also being developed, wherein liquid waste, converted to small glass granules about 3 mm in diameter, is incorporated in a metal matrix. A British process named FINGAL has been tested with waste canisters 15 cm diam by 1.5 m long. In another process, HARVEST, liquid waste and glass former are fed directly into a heated canister where drying, calcination and melting occur at 1030 to 1050 degree C. It is estimated that 3 parallel 30 cm diameter melters could vitrify the waste from a 5 ton/day fuel reprocessing plant. Similar processes are being developed in the U.S. The major emphasis of most processes has been in a vitrified form because of its high degree of nondispersibility and safety. (CAJ)

Waste Volume; Density

CALCINATION; FUEL REPROCESSING; WASTE TREATMENT; WASTES, LIQUID; WASTES, HIGH-LEVEL; SOLIDIFICATION; VITRIFICATION; VOLUME REDUCTION; WASTE PROCESSING; CALCINE; SILICATES; WASTES, RADIOACTIVE; WASTE STORAGE; WASTE VOLUME; WASTE MANAGEMENT; WASTE DISPOSAL; GLASS

<581>
Zarenkov, R.N., and A.A. Khonikevich.

Feasibility Study of Underground Storage of Bitumen Blocks Containing Radioactive Wastes.

HS-168-78; 7 pp.; Gigiena i Sanitariya 38(9):55-57. (1973, September)

The degree of danger from storage of bitumen blocks buried in soil with a high water table is evaluated. In the area of interest, the formations are the result of marine, lake-glacier, and glacier sand deposition. The soil and vegetation layer is up to 0.2 m and the layer below is a marine medium granular sand 0.7 to 2.6 m thick. Under the sand layer are clays at the 3.2 to 8.8 m level. Ground water was found to occur from 0.2 to 5 m in depth. Of the three bitumen blocks, blocks 1 and 2 were placed in the saturated zone and blocks 3 the aerated zone. The blocks are 25 to 30 liters in volume and contained the following activities in Ci/kg: beta 5 X 10(E-5), Sr 90 2 X 10(E+5), Cs 137 2 X 10(E-6) for the first block; and beta 6 X 10(E-5), Sr 90 3 X 10(E-5), Cs 137 4 X 10(E-6) for blocks 2 and 3. On the last two blocks a 2 to 4 mm layer of pure bitumen was put on the outside. After six months of burial, the concentration of Sr 90 and Cs 137 did not exceed the average annual allowable concentration established for drinking water. The highest concentrations case from block 1, which did not have the bitumen coating. Calculations were done to estimate travel time of the isotopes. For Sr 90 it would take 280 years to travel 1 km and Cs 137 would take 800 years. During the travel times, Cs 137 would decay by a factor of 1.6 X 10(E+7) and Sr 90 by a factor of 10(E+3). Using the infiltration rate of 170 mm/year an estimate of contamination of ground water and dilution was made. It was found that the granular sand would allow contamination to leak and Sr 90 would be decreased by a factor of 22. From these experiments the researchers feel that burial of treated bitumen blocks is acceptable. (BDV)

A test of the leaching rates of bitumen stored below the water table.

Depth to Water Table; Stratigraphic Unit

<581> CONT.
Thickness; Mean Precipitation

Sr 90; Cs 137

BITUMENS; BURIAL; CONTAINMENT; CONTAMINATION; DRINKING WATER; FIELD STUDIES; FISSION PRODUCTS; GEOLOGY; GROUND WATER; INFILTRATION; LEACHING; SEDIMENTS; SOILS; WASTE STORAGE; WASTES, LIQUID; WASTES, SOLID

<582>
Campbell, R.B. (Ed.), Exxon Nuclear Company, Incorporated.

High-Level Radioactive Waste Management.

CONF-760346; Advances in Chemistry Series 153; Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1976. American Chemical Society, Washington, DC, 166 pp.(1976)

The book contains 12 chapters, each of which has been abstracted separately. These chapters range from the Atomic Energy Commission's philosophy concerning high-level radioactive waste management and descriptions of a number of AEC supported programs to commercial waste processing facilities and a review of high-level radioactive waste management in Europe. (JET)

WASTE MANAGEMENT; WASTE PROCESSING; WASTES, HIGH-LEVEL; WASTES, RADIOACTIVE; REVIEWS; LABORATORY STUDIES; FIELD STUDIES; WASTES, SOLID; SOLIDIFICATION; WASTE STORAGE; WASTES, LIQUID; FIXATION; CLAYS; PLUTONIUM; RADIONUCLIDE MIGRATION; ROCKS; CESIUM; STRONTIUM; FUEL ELEMENTS; TRANSURANICS

<583>
Chapin, J.A., and R.E. Mine, EG and G Idaho, Incorporated, Idaho Falls, ID.

Reprocessing Facilities, Idaho Chemical Processing Plant.

TREE-1250 (Volume 2); Decontamination and Decommissioning, Long Range Plan, Idaho National Engineering Laboratory, C.R. Tolman (Ed.), Appendix B, (pp. B-1 - B-80), 319 pp.(1978, June)

The activities at the Idaho Chemical Processing Plant (ICPP) are discussed. These activities are fuel reprocessing, fluidized-bed waste solidification and research and development. The ICPP is a multipurpose facility for recovering enriched uranium from a wide variety of spent nuclear reactor fuels. Generally, when enough of one type of fuel is accumulated it is dissolved in nitric or hydrofluoric acid, and uranium is recovered from the dissolver effluent by solvent extraction. The product from the final extraction cycle, uranyl nitrate solution, is converted to solid UO_3 in a fluidized-bed denitrator. High-level liquid wastes are stored in the ICPP tank farm before fluid-bed solidification and solidified high-level wastes from the Waste Calcining Facility are stored in the high-level solids storage facilities. Those waste streams requiring monitoring at the waste treatment plants are the high-level liquid raffinates from the extraction system, intermediate-level liquid waste, contaminated organic wastes, dissolver and vessel off-gases, and solid wastes contaminated with radioactivity. Intermediate-level liquid waste is treated in the process equipment waste evaporator and ion-exchange system. A decontamination factor of 10(E+4) is achieved by evaporation alone. High-level wastes are treated in the Waste Calcining Facility. Control on the liquid and gaseous effluents is mentioned. Storage and other radioactive materials are discussed. (RDV)

Numerous diagrams are presented.

ACIDS; CALCINATION; COMBUSTION; CONTAMINATION; DECONTAMINATION FACTORS; EVAPORATION; FUEL REPROCESSING; FLUIDIZATION; ION EXCHANGE; MONITORING; PLANTS, WASTE TREATMENT; REACTORS; TANKS; URANIUM; WASTE MANAGEMENT; WASTE STORAGE; WASTE TREATMENT; WASTES, GASEOUS; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, SOLID; SOLIDIFICATION; ENCAPSULATION

<584>
Choi, E.C., T.S. Drolet, W.R. Stewart, and R.V. Campbell, Ontario Hydro, Toronto, Ontario, Canada.

Operation of Low-Level Radioactive Waste Incinerator.

CONF-790819; Proceedings of the 15th DOE Nuclear Air Cleaning Conference, Boston, MA, August 1978, (pp. 70-81).(1979)

Ontario Hydro's radioactive waste incinerator designed to reduce the volume of low-level combustible wastes from nuclear generating stations was declared in-service in September 1977. The wastes processed in this system may contain cotton and plastic materials, gloves, cardboard, paper, wood, oil absorbers, etc. with a specific activity on the order of 10(E-4) Ci/cu m. Up-to-date about 1500 cu m of combustible waste have been processed in over 90 separate batches. The incinerator's design capacity is 17 cu m/burn (600 cu ft/burn), and about 2200 kg of wastes can be processed in each burn cycle. The process has resulted in a 40:1 reduction in the volume and a 12.5:1 reduction in the weight of the Type 1 wastes. The ultimate volume reduction factor after storage is 23:1. Airborne emissions have been well below 1% of the Derived Emission Limits. Operating experience has proven that incineration is a viable process to reduce the volume of wastes stored, and thus the storage cost. (RAP)

Waste Volume

I 131; Ce 144; Cr 51; Cs 137; Zn 65; Co 60; Zr/Nb 95; Ru 106; H 3

WASTES, LOW-LEVEL; WASTES, SOLID; INCINERATION; VOLUME REDUCTION; EMISSIONS; PARTICLES, AIRBORNE; GASES; ASHES; PARTICLE SIZE; MAXIMUM PERMISSIBLE CONCENTRATION; COBALT 60; RUTHENIUM 106; CESIUM 137; CESIUM 134; CERIUM 144; ZINC 65; TRITIUM; IODINE 131

<585>
Claiborne, H.C., Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

<585> CONT.
Effect of Actinide Removal on the Long-Term Hazard of High-Level Waste.

ORNL/TM-4728; 44 pp. (1975, January)

A possible method for improving the management of high-level radioactive waste involves the removal of actinides to a very high degree to reduce the long-term hazard potential of wastes stored in geological formations. In this method, actinides are removed to levels where their contribution to the hazard potential is comparable to that from the long-lived fission products. Removal of actinides to this extent reduces the long-term (>1000 years) hazard potential of high-level waste to about 5% of that for pitchblende mineral on the basis of the hazard index. Calculations show removals of 99.99% plutonium, 99.9% uranium, americium, and curium, and 95% neptunium from spent uranium-plutonium fuels accomplish this result. High Temperature Gas-Cooled Reactor fuel will also require removal of 99.5% thorium and 95% protactinium, and increasing the uranium removal to 99.99%. Attaining these actinide removal percentages will require removal factors that are 10 to 100 times greater than have been achieved with existing high-radiation-level processing technology. Consideration of removal of long-lived fission products and additional actinides does not appear to be warranted until it has been demonstrated that actinides can be removed to levels approaching the present calculated values. These values of actinide removals serve as a desirable goal in process development studies of the waste fractionation program at Oak Ridge National Laboratory. (Auth)

The percentage contribution of individual actinide components to the hazard of waste of various types of fuels is given in tables.

ACTINIDES; SEPARATION PROCESSES; FRACTIONATION; THEORETICAL STUDIES; CALCULATIONS; WASTES, HIGH-LEVEL; HAZARD ANALYSIS; FISSION PRODUCTS; PLUTONIUM; URANIUM; AMERICIUM; CURIUM; NEPTUNIUM; SPENT FUELS; REACTORS, HIGH-TEMPERATURE GAS-COOLED; THORIUM; PROTACTINIUM

<586>
Cranfall, J.L., Savannah River Laboratory, Aiken, SC.

U.S. Program for the Immobilization of High-Level Nuclear Wastes.

DP-MS-79-2X; CONF-790602; American Nuclear Society Spring Meeting, Atlanta, GA, June 3-8, 1979, (3 pp.).(1979)

The multibarrier philosophy calls for the immobilization of radioactive wastes in non-dispersable, insoluble solid form as the first step in any disposal system. Current quantities of U.S. high-level wastes include 580 X 10(E+6) Ci of alkaline salt/sludge/liquor and separated Sr 90/Y and Cs 137 Ba at Hanford; 70 X 10(E+6) Ci of acid calcine and acid liquid at the Idaho Chemical Processing Plant; 560 X 10(E+6) Ci of alkaline salt/sludge/liquor at the Savannah River Plant; 64 X 10(E+6) Ci of alkaline salt/sludge/liquor at Nuclear Fuel Services; and over 1900 X 10(E+6) Ci of acid liquor and spent fuel from LWR's. If commercial wastes are reprocessed, their wastes can presumably be generated in the form best adapted for immobilization. However, for existing defense wastes, waste removal and pre-immobilization treatment is expected to be much more expensive than the preparation of the immobilization forms themselves. The earliest that full-scale immobilization could start is 1984, and the earliest that existing wastes could be fully immobilized is 2000. Thus, it is essential to get started as soon as possible on engineering design, but sufficient time is still available for a full research assessment before any irrevocable steps are taken. Accordingly, the proposed program consists of three interrelated parts: (1) development and NEPA choice between the major management alternatives at existing waste sites (in situ, on site, or off site disposal); (2) waste form development; and (3) waste form characterization. A large number of alternate forms have been proposed for waste immobilization. These include calcine, rich clay, normal concrete, hot pressed concrete, pellitized calcine, glass, clay ceramic, supercalcine, synroc, glass ceramic, pellet in metal matrix, coated supercalcine in metal matrix, and Cermet. Most of the emphasis to date has been on glass. (LKM)

WASTES, SOLID; WASTES, HIGH-LEVEL; WASTES, LIQUID; SOLIDIFICATION; IMMOBILIZATION; SALTS; SLUDGES; CALCINE; SPENT FUELS; DESIGN; WASTE MANAGEMENT; GLASS; VITRIFICATION; REVIEWS

<597>
Dau, G., and R. Williams, Battelle-Pacific Northwest Laboratories, Richland, WA; Electric Power Research Institute, Nuclear Power Division, Nuclear Systems and Materials Department, Palo Alto, CA.

Secure Storage of Radioactive Waste.

Aware, October 1976:6-11.(1976, October)

Possible solutions to the questions of safe storage and transport are explored. For high-level wastes conversion to a solid and immobilization in glass is the most promising system. The steps involved are: fuel reprocessing, short-term high-level liquid storage and two-step solidification, transportation and final disposal. Technology is currently available for all the above steps. High-level waste comes into being when uranium and plutonium are separated by solvent extraction from the fission products, 99% of which come off as a concentrated acidic aqueous stream. One metric ton of uranium ultimately produces 1100 gallons of HLW and by the year 2000, 29 million gallons will be produced by 625,000 MWe nuclear capacity. The short-term storage tanks are double walled stainless steel tanks inside a concrete vault. To reduce the volume of waste the liquid is calcined and according to NRC regulations the liquid must be in solid form 5 years after reprocessing. The calcined material occupies one-eighth the volume of the liquid waste. Once calcined the waste may then be vitrified. The glass produced is 1000 times less likely to disperse radionuclides to the environment than is the calcine. Radiation effects tests have shown that glass is a durable and reliable medium to immobilize the waste. Work is underway to make calcination and vitrification a one-step process. Transporting glass in shielded canisters will allow a negligible effect on the general

<587>

WASTE PRODUCTION

<587> CONT.
public and workers. Final disposal will be in deep underground cavities remote from the local ground water table. Further work in waste management is being done by the responsible government agencies. (NDV)

Waste Volume

CALCINATION; CONTAINER INTEGRITY; CONTAINERS; FUEL REPROCESSING; IMMOBILIZATION; GLASS; SHIELDED CONTAINERS; TANKS; WASTE MANAGEMENT; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, LIQUID; VOLUME REDUCTION; VITRIFICATION; TRANSPORTATION; WASTE VOLUME; VOLUME REDUCTION

<588>
Dillon, R.L., B. Griggs, R.S. Kemper, and R.G. Nelson, Battelle-Pacific Northwest Laboratories, Richland, Wh.

Chemical Decontamination and Melt Densification of Chop-Leach Fuel Hulls.

CONF-760310; STI/PUB/433; IAEA-SM-207/69; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 1, (pp. 185-199), 397 pp. (1976)

This paper summarized progress at Battelle-Pacific Northwest Laboratories on decontamination and densification of fuel bundle residues. Most reprocessing involves dissolution of spent UO2-PuO2 fuel from the sheared fuel bundle, leaving a residue of Zircaloy, stainless steel, and Inconel fragments contaminated by transuranics. From the standpoint of waste stream minimization and ease of operation, the major decontamination problem is removal of the zirconium dioxide corrosion product layer from the surfaces of Zircaloy fragments. An aqueous decontamination process, using HF at 550-600 degrees C to undercut and aqueous reagents (ammonium oxalate, ammonium citrate, ammonium fluoride, and hydrogen peroxide) to remove the layer, appears to be very effective. A possible alternative which would avoid side reactions of zirconium with water would be to use various gaseous fluorine compounds. Treatment with molten fluoride salts also proved effective, but produced a large metal loss and salt drayout. TRU contamination from trump uranium in Zircaloy is not removed by surface cleaning and may require molten metal decontamination methods. Melting of the solid fuel residues in graphite crucibles produced a low-melting alloy of Zircaloy, stainless steel, and Inconel; Zircaloy cladding sections melted separately by the Inductoslag process have yielded ingots suitable for storage. If the fuel stock is descaled before melting, the Zircaloy ingots may be used for refabrication. The absorption rate and equilibrium pressure of hydrogen over Zircaloy and Zircaloy, stainless steel, Inconel alloy ingots were studied at various gas pressures and concentrations. Both types of ingots absorb hydrogen rapidly enough to be valuable for storage of hydrogen and tritium released by the melting of fuel residues. (LKN)

Pu 241

ACTINIDES; CONCENTRATION; DECONTAMINATION; EXTRACTION; FUEL REPROCESSING; LABORATORY STUDIES; METHODS; REPROCESSING; RETRIEVABILITY; TRANSURANICS; TRITIUM; VOLUME REDUCTION; WASTE MANAGEMENT; WASTE TREATMENT; WASTES, RADIOACTIVE; WASTES, SOLID; WASTES, TRANSURANIC; ZIRCALOYS; COBALT; PLUTONIUM; ALLOYS

<589>
Duckworth, J.P., Nuclear Fuel Services, Inc., West Valley, NY.

The High-Level Radioactive Waste Management Program at Nuclear Fuel Services.

CONF-700346; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, R.E. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, Ca, April 1-2, 1970. American Chemical Society, Washington, DC, Ch. 5, (pp. 72-83), 166 pp. (1976)

The Nuclear Fuel Services plant at West Valley, New York, was designed to use the standard Purex and Thorex processes with two liquid waste streams: (a) low-level liquid waste such as process condensates and cooling water containing only trace quantities of radioactivity which could be discharged directly into the surrounding waterways, and (b) high-level liquids. The Purex-type waste represents combined high-level and intermediate-level waste neutralized and stored in multiconfined mild steel tanks. The Thorex waste, which also contains the thorium is stored as an acid solution in stainless steel storage vaults. Future high-level waste management systems include multitank complex for five year cooling and storage as an acidic solution. The location, structure, and multiple containment provides protection against possible releases to the environment. Sketches are provided in this chapter showing the storage systems and tables of waste compositions are included. (Auth) (JWT)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; TANKS; WASTE STORAGE; PUREX PROCESS; THOREX PROCESS; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTE MANAGEMENT

<590>
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Neutron and Gamma-Ray Sources in Light Water Reactor High-Level Nuclear Waste.

SAND-77-0369; 18 pp. (1977, June)

Predictions of the composition of high-level waste from U-fueled LWRs have been used to calculate the neutron and gamma-ray sources in such waste at cooling times of 3 and 10 years. The results are intended for interim application to studies of waste shipping and storage pending the availability of more exact knowledge of fuel recycling and of waste concentration and solidification; they are subject to large uncertainties and should be treated as estimates only. The highest actinide activities in HLW from 1 MT of LWR U-fuel after reprocessing and 3 years of decay are from Np 239 (10.2 Ci), Pu 238 (96.9 Ci), Pu 241 (866 Ci), Am 241 (160 Ci), Am 243 (18.2 Ci), Cm 242 (168 Ci), and Cm 244 (2160 Ci); after 10 years of decay, the dominant actinides are Np 239 (18.2 Ci), Pu 238 (92.9 Ci), Pu 241 (320 Ci), Am 241 (163 Ci), Am 243

<590> CONT.
(19.2 Ci), and Cm 244 (1640 Ci). The highest fission product activities after 3 years of decay are from Sr 90 (71,300 Ci), Y 90 (71,300 Ci), Ru 106 (50,800 Ci), Rh 106 (50,800 Ci), Cs 134 (77,000 Ci), Cs 137 (99,600 Ci), Ba 137m (93,100 Ci), Ce 144 (51,900 Ci), Pr 144 (51,900 Ci), and Pm 147 (44,000 Ci); after 10 years, the dominant fission products are Sr 90 (60,000 Ci), Y 90 (60,000 Ci), Cs 134 (7220 Ci), Cs 137 (84,700 Ci), Ba 137m (79,200 Ci), Pm 147 (6900 Ci), Sm 151 (1150 Ci), and Eu 154 (8450 Ci). Gamma ray intensities from fission product sources in this waste after 3 years based on energy groups range from 2.8 X 10(E+11) to 1.6 X 10(E+5) photons/cu cm sec; after 10 years the values are 6.8 X 10(E+10) - 1.1 X 10(E+3) photons/cu cm sec. The major contributing isotopes are Rh 106, Pr 144, Y 90, Cs 134, Eu 154, Ba 137m, Ce 144, and Cs 137. Actinide contribution to the total gamma ray source is negligible, and the values for fission products may be considered to be the total source. Cm 244 is the primary contributor to neutron production, with a spontaneous fission neutron production rate of 5.1 X 10(E+3) n/cu cm sec after 3 years of decay, and 3.9 X 10(E+3) n/cu cm sec after 10 years. After 3 years of decay, 0.18(n,n)F 21 neutron production from Cm 244 is 42 n/cu cm sec; after 10 years, it is 32 n/cu cm sec. (Auth) (LEH)

WASTES, HIGH-LEVEL; REACTORS, LIGHT WATER; NEUTRONS; RADIATION, GAMMA; RADIATION SOURCES; TRANSPORTATION; WASTE STORAGE; RADIONUCLIDES; RECYCLING; FUEL CYCLES; ACTINIDES; NEPTUNIUM 239; PLUTONIUM; AMERICIUM; CURIUM; FISSION PRODUCTS; STRONTIUM 90; YTTRIUM 90; RUTHENIUM 106; RHODIUM 106; CESIUM; BARIUM 137M; CERIUM 144; PRASEODYMIUM 144; PROMETHIUM 147; EUROPIUM 154; SAMARIUM 151; THEORETICAL STUDIES; WASTE MANAGEMENT; PREDICTIONS

<591>
Ewest, E., Hahn-Meitner-Institut fur Kernforschung, GmbH, Berlin, German Federal Republic.

Calculations of Radioactivity Release Due to Leaching of Vitrified High Level Waste.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 161-168), 563 pp. (1979)

A model for the calculation of radioactivity release due to leaching from vitrified waste is presented which uses empirical equations with coefficients depending on parameters such as glass and waste composition, flow rate of the leachant, temperature, pH, and geometry of the sample. The following assumptions are made for the mass transfer equations: homogeneous leaching of the glass; glass surface area is constant up to a mass transfer of 25% of the sample; integrity of the solidified product is preserved; and leaching data obtained for glass particles are valid for glass blocks. Corrosion, diffusion and a combination of both are the kinetic equations considered in a computer program. The results indicate the radioactivity available from unsolidified waste decreases from 10(E+4) to 500 within the first 10(E+3) years. After that time the safety margin provided by immobilization decreases to about 10 after 10(E+5) years. (NDV)

WASTES, HIGH-LEVEL; GLASS; EQUATIONS; COMPUTER PROGRAMS; LEACHING; TEMPERATURE; pH; IMMOBILIZATION; CORROSION; DIFFUSION

<592>
Fosse, R.L., and W.T. Gregory, III, Lowa Corporation, Charlotte, NC.

Volume Reduction of Liquid Radioactive Wastes Using Mechanically Agitated Thin-Film Evaporators.

American Institute of Chemical Engineers, Proceedings of the 85th National Meeting, Philadelphia, PA, June 4-8, 1978, (37 pp.). (1978, June)

The potential and ability of mechanically agitated thin-film evaporators to reduce the volume of liquid radwaste at nuclear power plants is discussed. Specifications for a variety of evaporators as well as maintainance is presented. With the waste streams of both types of light water reactors producing from 76,000 gal/day to 365,000 gal/day volume reduction is necessary. A radwaste evaporator may concentrate the wastes to evaporator bottoms containing 25% total solids. Some designs however, may only concentrate to the level of 10 to 15% before fouling or encrustation occurs. An evaporator-crystallizer system has an advantage over most systems because it can reduce the waste by about 50%. The cost for the first year of operation of a radwaste facility may run as high as $11.2 million, with the cost running about 1.5 million for the succeeding years. Hence, it is necessary to have volume reduction. Several sample problems of waste concentration are illustrated in detail. Thin-film evaporator reliability is demonstrated, and encapsulation in bitumen and polymers is briefly discussed. (NDV)

BITUMINIZATION; DECONTAMINATION; ECONOMICS; EVAPORATION; EVAPORATORS; PLANTS, WASTE TREATMENT; POLYMERS; REACTORS, LIGHT WATER; VOLUME REDUCTION; WASTE MANAGEMENT; WASTE TREATMENT; WASTES, LIQUID; WASTES, RADIOACTIVE; SOLIDIFICATION; ENCAPSULATION

<593>
Feher, L., J. Hirling, and J. Kasza, Metropolitan Public Health and Epidemiology Station, Budapest, Hungary; Isotope Institute of the Hungarian Academy of Science, Budapest, Hungary.

Technological Facilities at the New Hungarian Radioactive Waste Handling and Disposal Station.

Izotoptechnika Budapest 20:74-87; OLS-78-99. (1977; 1978)

The specifications of a new Hungarian waste handling and storage plant are fully discussed. It conforms to the established guidelines for installations which present similar radioactive risks in regard to geological, environmental, public health, and technological factors. Radioactive wastes are collected from institutions using isotopes and are deposited in storage.

<593>

WASTE PRODUCTION

<593> CONT.

Approximately 60 cu m of liquid wastes are received at the plant at a row of siphoning booths. The waste is classified and purified as necessary by flocculation or ion exchange. The concentrates are then mixed with cement and stored. Liquid organic solutions are absorbed into diatomaceous earth or saponified before storage. Biological wastes are placed in melted asphalt. Solid wastes, 80 cu m annually, are collected in plastic bags or metal drums without further treatment and stored. Low and high-activity waste radiation sources are stored separately in special stainless steel pipe wells 6 m deep. (NDN)

Waste Volume

ABSORPTION; CEMENTS; COST BENEFIT ANALYSIS; DECONTAMINATION; FIXATION; FLOCCULATION; ION EXCHANGE; METHODS; PLANTS, WASTE TREATMENT; WASTE MANAGEMENT; WASTE STORAGE; WASTE TREATMENT; WASTES, LIQUID; WASTES, SOLID; WASTES, RADIOACTIVE; SOLIDIFICATION; ENCAPSULATION; ASPHALT

<594>

Fernandez, S.J., and R.C. Girton, Idaho National Engineering Laboratory, Idaho Falls, ID.

Methods Evaluation and Development for the Monitoring of Technetium 99 and Selenium 79 in Nuclear Fuel Reprocessing and Waste Solidification Facility Off-Gas.

ICP-1170; 27 pp.(1978, November)

The purpose of this study was to investigate and evaluate methods of monitoring gaseous Tc 99 and Se 79 which may be emitted from fuel reprocessing and waste solidification facilities. The proposed continuous technetium and selenium monitors consisted of a rapid wet chemical separation from beta emitting interferences. This continuous technetium monitor was based on the distillation of the incoming technetium as Tc2O7, drying the sample stream by permeation distillation, and recollection of the technetium on a silver impregnated filter. The collected technetium was counted by a reggedized surface barrier detector. The detection limit for 2-hr counting periods was 10(E-10) uCi/cu cm using a 20 cu cm counting chamber, sufficient for radiological protection, but inadequate for the Tc levels anticipated at the Idaho Chemical Processing Plant (ICPP) stack. A selenium continuous monitor was based conceptually on distillation from 20% HBr. The distillation yield was found to be adequate, but the counting efficiency of the surface barrier detector was insufficient for the lower energy Se 79 beta. A time-integrated sampling method that quantitatively collected gaseous Tc and Se was developed using silver zeolite as a collection medium. A technique for dissolution and chemical separation of the selenium and technetium from the silver zeolite matrix was developed, having chemical yields of 18% and 10%, respectively. The silver zeolite sampler, demonstrated on the ICPP stack, detected selenium-79 during waste solidification and technetium-99 during fuel dissolution. Although the concentrations observed were in the range of 10(E-10) - 10(E-11) uCi/cu cm, this was 5-10 times the detection limit for the time-integrated sampler. Stack gas concentrations of 10(E-10) uCi/cu cm correspond to site boundary levels, more than 8 orders of magnitude below present standards. The developed chemical separation procedure was compatible with liquid scintillation counting and free of interferences commonly found in the off-gas waste solidification facilities. These results indicate the time-integrated sampling technique is suitable for industrial use. (Auth) (KEN)

Tc 99; Se 79

TECHNETIUM 99; SELENIUM 79; GASES; MONITORING; EMISSIONS; WASTE TREATMENT; PLANTS, WASTE TREATMENT; RADIOACTIVITY; SEPARATION PROCESSES; CHEMICAL ANALYSIS; SAMPLING; METHODS; LABORATORY STUDIES

<595>

Fey, F.L., Northern States Power Company, Monticello, MN.

Operation of the Monticello Nuclear Generating Plant Liquid and Solid Radioactive Waste Disposal System.

Health Physics 23:438.(1972)

The Monticello Nuclear Generating Plant (a General Electric Boiling Water Reactor) has employed the following means and methods to minimize the discharge of liquid radioactive waste. The liquid radwaste system is rather unique in its use of a powdered ion exchange filter which is applied as a "precoat" over a network of supporting filter elements referred to as a "filter/demineralizer". The precoated element removes both the dissolved and suspended solids from the process stream. When the filter/demineralizer becomes exhausted due to either suspended solids (high pressure drop) or dissolved solids (high conductivity) the precoat is backwashed off of the supporting elements into a receiving tank. At some convenient time the liquid portion is separated from the solid portion by decantation and centrifugation. The liquid portion is processed back to the primary system and the solid portion is mixed with cement, placed in 55 gallon drums, and shipped to a burial facility. The condensate polishing system, reactor cleanup system, and fuel pool cleanup system also use a powdered ion exchange filter media. The radwaste system has been modified to the point where all floor drains, equipment drains, and laundry waste can be reclaimed; chemical waste is disposed of by using it in the cementing process. (Auth) (PAP)

Abstract is the complete text.

VOLUME REDUCTION; WASTES, LIQUID; ION EXCHANGE; FILTERS; WASTES, SOLID; CEMENTS

<596>

Fitzgerald, C.L., W.H. Godtee, and P.E. Blanco, Oak Ridge National Laboratory, Oak Ridge, TN.

The Feasibility of Incorporating Radioactive Wastes in Asphalt or Polyethylene.

Nuclear Applications and Technology 9:921-929,(1970, December)

Laboratory studies on the use of polyethylene as a substitute for asphalt and the

<596> COST.
application of both to organic and aqueous sodium borate wastes from nuclear power stations showed that for incorporation of organic wastes, a product composed of 35-38 wt% asphalt, 25 wt% TBP (the organic liquid used) and 37-40 wt% filler (fine clays, colloidal silica, and carbon black) was best. However, this material drained slowly from the evaporator and was soft at room temperature, so it was abandoned in favor of polyethylene. The best products of this type were 30 wt% TBP incorporated into DYLT polyethylene; leach rate (fraction leached/sq cm/day) was 1.5 X 10(E-5). 20-40 wt% evaporator concentrate salts and 20-40 wt% NaNO2 were each incorporated into DYLT and DYDT polyethylene; leach rates (fraction leached/sq cm/day) ranged from 1.6 x 10(E-4) to 3.1 X 10(E-4). All polyethylene products showed little or no alteration after irradiations up to 10(E+9) rads. Asphalts incorporating 50-60 wt% NaNO2 or 50-55 wt% Ca(NO2)2 displayed volume increases up to 35 and 70% under 10(E+8) and 10(E+9) rads, respectively. Flammability, shock sensitivity, and other tests demonstrated the safety advantages of the salts-polyethylene system. Polyethylene is recommended for disposal of organic wastes, and appears acceptable for oxidizing salts; asphalt is not recommended for wastes containing oxidizing salts because of the danger of fire. (LKH)

Leaching Rate

PLASTICS; ASPHALT; ORGANIC COMPOUNDS; EVAPORATOR CONCENTRATE; SALTS; SOLIDIFICATION; IMMOBILIZATION; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; RADIATION EFFECTS; WASTES, ORGANIC; LEACHING

<597>
Forberg, S., T. Westermark, E. Larker, and B. Nidell, The Royal Institute of Technology, Department of Nuclear Chemistry, Stockholm, Sweden; Central Research and Development Department, ASEA, Vasteras, Sweden.

Synthetic Rutile Microencapsulation: A Radioactive Waste Solidification System Resulting in an Extremely Stable Product.

CONF-791121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 201-205), 563 pp. (1979)

The development of a new process for synthesis of coarse-grained titanate ion exchangers is discussed. This is an effort to incorporate HLW into minerals known to be stable for long periods of time or synthetic materials simulating the desired condition. The ion exchange capacity is 2.5 eq/kg and the anticipated cost is low. For strontium the decontamination factor is about 2 x 10(E+7) and for cesium the decontamination factor is approximately 10(E+6). A few samples of the exchanger were sintered by hot isostatic pressing at 1,300 degree C and 150 MPa for 2 hours. Hardness and strength of the resulting material are high; the densities are higher than rutile, 4.26 g/cu cm; no porosity was observed under the microscope. Leaching tests indicated Cs would be lost at a rate of 1 cm/10(E+6) years. Yet to be studied are the effects of cracking and metamictization. These will be minimized. (BDV)

Ion Exchange Capacity; Density

SOLIDIFICATION; DECONTAMINATION FACTORS; ION EXCHANGE; ION EXCHANGE CAPACITY; DENSITY; LEACHING; POROSITY; TITANIUM COMPOUNDS; RUTILE; NETRONS; LABORATORY STUDIES

<598>
Swanell, L., Brookhaven National Laboratory, Upton, Long Island, NY.

The Brookhaven Experience with Sea Disposal of Radioactive Wastes.

Proceedings of the Sanitary Engineering Conference, Cincinnati, OH, December 6-9, 1955, (pp. 153-161). (1956)

This report summarizes procedures carried out by Brookhaven National Laboratory in the 1950's for the sea disposal of radioactive wastes. Low-level liquid and solid wastes are buried in the soil on-site; radioactive liquids up to 3 X 10(E-12) Ci/ml are dumped into the sewage system. Intermediate and high-level wastes [10(E-4)-10(E-5) Ci/ml] are evaporated to slurry if liquid, or baled if solid combustibles, and placed in concrete-lined 55 gal drums (lining is 2-8 in. thick). Other waste handled in this way includes activated metals and equipment, and contaminated sand from hot scrubbing operations. Hot slurry is mixed with concrete and added to the other materials packing the drums. The drums are sealed at only one end, and are transported to the disposal site inverted. The accumulated waste (600-700 drums) is transported once a year to Floyd Bennett Field by truck convoy, and from there by Navy LST to a site about 150 mi. offshore, where it is dumped in about 7500 ft. of water. All personnel exposures during the past two such shipments have been less than the weekly limits of 300 milliroentgens. Cost of disposal of last year's wastes (660 drums containing about 600 Ci) was about $20/drum. (LKH)

SEA DISPOSAL; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, SOLID; WASTES, LIQUID; CONCRETES; DRUMS; EVAPORATION; COST BENEFIT ANALYSIS

<599>
Godbee, H.W., Oak Ridge National Laboratory, Oak Ridge, TN.

Characteristics of Wastes.

ORNL-4751; Radioactive Waste Repository Project: Technical Status Report for Period Ending September 30, 1971, (pp. 135-145), (1971)

Characteristics of solid wastes containing plutonium and other transuranic elements are outlined. The wastes include such contaminated items as paper, cloth, wood, plastic, glass, ceramic and metal as well as salts and sludges from liquid waste streams and offgases. The most important isotopes encountered are Pu 238, Pu 239, Pu 240, Pu 241, and Am 241. The densities of the wastes range from 2 to 200 lb/cu ft, and up to two-thirds of the wastes are combustible.

<599> CONT.

Also up to three-fourths of the waste by volume can be reduced in volume by factors of 2 to 10 by compaction. Since these alpha wastes do not generate penetrating radiation, remote handling techniques are not required. Initially the repository will not take combustible alpha wastes, but a feasibility study was done on storing them. After several combustion tests the general conclusions are: combustible wastes packed in combustible containers should not be accepted at the repository and combustible wastes packed in combustible metal drums are acceptable at the repository. Weight losses in the combustion tests were lower when CaO was mixed with the waste. Losses ran 0.2 to 3.3% with the CaO and 6 to 12% without the CaO. High-level wastes generated during the processing of spent reactor fuels are converted into solid form by four processes. The fluidized-bed waste calcining facility in Idaho has converted 2 million gallons of wastes in to granular solids. The solidification of wastes at the Waste Solidification Engineering Prototypes at Battelle-Northwest demonstrated the pot, spray and phosphate glass processes. More than 50 million Ci of fission products were processed. For wastes to be accepted at the repository they must meet certain criteria. Liquid alpha wastes or potentially explosive chemicals, pyrophoric materials, oxidants and gas under pressure must be converted to solids as does the high-level wastes. (NDV)

Density

Pu 238; Pu 239; Pu 240; Pu 241; Am 241

ALPHA PARTICLES; CALCINE; COMPACTION; COMBUSTION; FISSION PRODUCTS; PACKAGING; PLUTONIUM; AMERICIUM; REPOSITORY; VOLUME REDUCTION; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, LIQUID; WASTES, SOLID; WASTES, TRANSURANIC; WASTE VOLUME

<600>

Golchert, N.W., and J. Sedlet, Argonne National Laboratory, Argonne, IL.

Determination of Transuranics in Environmental Water.

CONF-760386; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, R.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 10, (pp. 152-157), 166 pp. (1976)

Facilities that process high-level radioactive material are required to demonstrate that their radioactive emissions are below standards set by regulatory agencies. A sequential procedure for the separation of the transuranic elements from water samples up to 60 liters is discussed. The procedure involves the coprecipitation of the transuranic nuclides on calcium fluoride from acid solution after reduction of the plutonium and neptunium with bisulfite. The calcium fluoride precipitate is dissolved in aluminum nitrate-nitric acid solution and the plutonium and neptunium separated on an ion-exchange resin column. The column effluent solution is extracted to separate the transplutonium nuclides. The separated fractions are electrodeposited and the concentrations determined by alpha spectrometry. The chemical recoveries are determined by adding known amounts of plutonium 236, neptunium 239, and americium 243 for the transplutonium nuclides. (NRT)

PLUTONIUM; AMERICIUM; NEPTUNIUM; TRANSURANICS; WASTES, HIGH-LEVEL; EMISSIONS; CHEMICAL ANALYSIS; ION EXCHANGE; MONITORING; RESINS; PRECIPITATION, CHEMICAL; ELECTRODEPOSITION; SPECTROMETRY; CONCENTRATION; WATER

<601>

Goldstein, N., T. Gangwer, C. Braun, and R. Lester, Brookhaven National Laboratory, Department of Applied Science, Upton, NY.

Technology Assessment and Impact Analysis of Separation Methods Applied to Radioactive Waste Management.

BNL-50628; 33 pp. (1977, March)

A computer program has been written to estimate actinide production and accumulation in the high-level waste generated by an expanding nuclear economy over a 100-year period. Three options for treating the high-level waste are considered: 1) no partitioning, i.e., a continuation of current practices; 2) partitioning of the high-level and separate storage of the actinides; and 3) partitioning of the high-level waste and recycling of the actinides in light-water reactors. A preliminary assessment was carried out to determine the role of photochemistry in handling nuclear materials. A new photochemical method is described for separating actinides from high-level waste or from each other based on selective ligand exchange reactions. A preliminary cost comparison of Pu separation using standard photoredox techniques has been accomplished and is reported. Even though cost estimates appear very favorable, the suitability of photochemistry for commercial spent fuel reprocessing awaits verification by process testing. The energy requirements for photochemically separating all of the actinides are estimated to be only a fraction of the energy supplied by the fuel. The potential advantages and disadvantages of actinide partitioning as a waste management option are discussed. Conclusions and recommendations are made regarding both the direction of technical development and policy analysis. (NAT)

COMPUTER PROGRAMS; ACTINIDES; WASTES, HIGH-LEVEL; SEPARATION PROCESSES; COSTS; PLUTONIUM; PARTITIONING; THEORETICAL STUDIES; IONS; ABSORPTION; LASERS; PHOTOCHEMISTRY; WASTE STORAGE; REDOX REACTIONS; EQUATIONS; PHOTONS; FISSION PRODUCTS; RECOMMENDATIONS

<602>

Gombert, D., N.S. Cole, and J.R. Berreth, Idaho National Engineering Laboratory, Idaho Falls, ID.

Vitrification of High-Level ICPP Calcined Wastes.

ICP-1177; 31 pp. (1979, February)

Long-term storage of high-level wastes may require a mechanically nondispersible and chemically leach resistant form. One method

<602> CONT.

under consideration is to vitrify calcined waste into a leach-resistant glass which would contain 25 to 40% calcine. A full-scale process to vitrify defense waste at the Idaho National Engineering Laboratory (INEL) could process all of the existing high-level calcined waste and that produced in a 10 year period after process initiation. The process would vitrify zirconia, alumina, stainless-steel, and Zr-Mo blend calcines produced at the Idaho Chemical Processing Plant (ICPP). Glasses of 116 compositions were made with simulated zirconia waste and compared on the basis of viscosity at the melting temperature and gross weight loss in an acid leach. The glasses were primarily borosilicates with minor additives (1-5 wt%) of Li2O, B2O3, CaO, and ZnO to the SiO2, Na2O, Al2O3 matrix. The experimental glasses compared many combinations of these additives at varying concentrations and at melt temperatures of 1050 to 1200 degrees C, and waste loadings of 25 to 40 wt% calcine. Glasses had a density of 2.5 to 2.75 g/cm cm and softened at about 520-570 degrees C. Thermal conductivities over 310-490 K were 0.83-0.74 W/m-K and porosities were less than 5 vol%. Fluoride volatility was 5 to 20% at the 1100 degrees C melt temperature. The crystalline content ranged from 1 to 8 wt% and consisted primarily of CaF2. Crystalline CaF2 concentration decreases over at least the first 2 weeks when the glass is kept at 700 to 800 degrees C, but other crystalline forms appear as the glass devitrifies. Soxhlet leach rates did not change significantly. The most promising glass on the basis of low melt viscosity at 1100 degrees C and high leach resistance is a glass containing 44% SiO2, 16% Na2O, 5.33% B2O3, 1.33% CaO, and 33.34% zirconia calcine. This glass (No. 54) can also incorporate variations up to 20% in alumina and up to 15% dolomite (a calcining startup material) with no significant deterioration of quality. Further testing of effects of devitrification on leach resistance and mixed alkali and Fe2O3 effects on viscosity and leach resistance are underway. (Auth) (LEM)

Details of experimental glass formulations and properties are provided in two appendices.

Density; Temperature; Thermal Conductivity; Porosity

VITRIFICATION; GLASS; CALCINATION; CALCINE; WASTES, HIGH-LEVEL; ZIRCONIUM; VISCOSITY; LEACHING; BORON COMPOUNDS; SILICATES; DENSITY; LABORATORY STUDIES; CHEMICAL COMPOSITION

<603>
Goodlett, C.B., Savannah River Laboratory, Aiken, SC.

Getting Evaporators to Work: Concentration of Aqueous Radioactive Waste.

Chemical Engineering Progress, April 1976:63-66.(1976, April)

Vertical and horizontal wiped film evaporators were tested at the Savannah River Plant for concentration of aqueous Purex waste to a product that would be solid at room temperature. The tests showed that: (1) waste storage tanks can be filled to 92% capacity, using three evaporators per tank, based on the slope of the solidified salt cake (8 degrees); (2) solids did not accumulate on the heat transfer surfaces of the evaporators; (3) average heat transfer coefficient was 200 BTU/hr-sq ft-degree F; (4) the salt cake product contained less than 20 wt% water, including approximately 10% combined water; (5) entrainment of product in the process condensates was 1 part in 10,000-20,000 based on sodium ion analyses; (6) mechanical problems were readily overcome by minor design changes; (7) corrosion and wear were minimal. Tests have also shown that HB waste and decladding waste can be similarly concentrated. Based on these results, two prototype vertical evaporators were designed, to be installed in access ports of existing waste storage tanks. (LEM)

EVAPORATION RATE; CONCENTRATION; EQUIPMENT; EVAPORATION; EVAPORATORS; EVAPORATOR CONCENTRATE; SALTS; WASTE TREATMENT; WASTES, RADIOACTIVE; WASTES, LIQUID; WASTE PROCESSING; DECLADDING; WASTES, HIGH-LEVEL; TANKS; PUREX PROCESS; HEAT TRANSFER; WASTE MANAGEMENT

<604>
Gray, W.J., Battelle-Pacific Northwest Laboratories, Materials Department, Ceramics and Graphite Section, Richland, WA.

Volatility of a Zinc Borosilicate Glass Containing Simulated High-Level Radioactive Waste.

BNWL-2111; 28 pp.(1976, October)

Volatilities of a zinc borosilicate glass containing 25 wt % simulated high-level radioactive waste calcine and of the calcine itself were determined in the range 800-1200 C. The fission products Rb, Mo, Ru, Te, and Cs were detected in the vapor from both the glass and the calcine. In general, the vaporization rates were lower from the glass with the exception of Rb and Cs at 1200 C, where the rates were about the same from the glass as from the calcine. Furthermore, slopes of Arrhenius plots for Rb and Cs were higher for the glass indicating that more of these elements would be lost from the glass above 1200 C than from the calcine. Addition of moisture to the air significantly increases the vaporization rate for the glass. Heats of vaporization of Cs and Rb from calcine are very low (16.7 and 15.7 kcal/mole, respectively), a fact that is not understood. Release rates for Cs and Ru determined in this work are 1.5%/hr at 100 C and 33%/hr at 1200 C for Cs, and 0.07%/hr at 1000 C and 1.3%/hr at 1200 C for Ru. Vapor pressure is conservatively estimated to be less than 1 atm for temperatures below 1800 C for the glass and 1700 C for the calcine. Vaporization of all volatile elements from the glass seems to be dominated by a single mechanism; thus, vapor composition over the glass is essentially temperature independent. Several mechanisms apply to vaporization of the various elements from the calcine. Gross weight-loss from the glass and calcine appears to be surface and diffusion controlled, respectively. (Auth) (LEM)

Heat of Vaporization

Cs; Ru

GLASS; LABORATORY STUDIES; CALCINE; FISSION PRODUCTS; VOLATILITY; WASTES, HIGH-LEVEL; TEMPERATURE; DIFFUSION; VAPORIZATION; WASTE MANAGEMENT; WASTE TREATMENT; BORON COMPOUNDS; LABORATORY STUDIES; CORROSION

<605>

WASTE PRODUCTION

<605>
Gauber, H., H. Hussain, E. Kahl, G. Ondracek, and J. Saidl, Kernforschungszentrum, Karlsruhe, Federal Republic of Germany.

Preparation and Characterization of an Improved High Level Radioactive Waste (HAW) Borosilicate Glass.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, C.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 37-42), 563 pp.(1979)

A glass composition based on the borosilicate system with TiO2 was developed and evaluated for the fixation of high-level radioactive waste. The TiO2 was used to minimize the volatility of cesium during melting. Glass frit and denitrated inactive simulated HAW solution was melted to produce the glass product. The parameters evaluated were: specific gravity; leach resistance against distilled water; linear expansion; viscosity of melt; impact resistance; thermal conductivity; electrical conductivity; volatility of the inactive components during melting; and recrystallization behavior. More studies are in progress on two types of glasses evaluated. Glass products with real HAW from LWR and LMFBR fuel reprocessing were investigated. In two years no structural change in the glass products influenced by irradiation has been detected. (NDV)

CRYSTALLIZATION; FIXATION; ENCAPSULATION; LABORATORY STUDIES; GLASS; PHYSICAL PROPERTIES; SILICATES; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, LIQUID; METHODS

<606>
Haden, W.L., Jr.

Development of an Effective Aluminosilicate Adsorbent for Radioactive Waste Disposal.

TID-7640; Use of Inorganic Exchange Materials for Radioactive Waste Treatment, D.K. Jamison, et al. (Eds.), Proceedings of a Working Meeting, Washington, DC, August 13-14, 1962. U.S. Atomic Energy Commission, Division of Technical Information, Washington, DC, (pp. 201-210), 238 pp. (TID-7640).(1963, January)

A need for uniform clinoptilolite was alleviated by pelletizing. The combination of colloidal attapulgite and sodium hydroxide produced stable pellets. In order to test the stability, the pellets were boiled for 10 minutes in water and 20% HNO3. The attapulgite allowed the mixture to be readily extrudable and the calcium and magnesium content precipitated the silicates as active binders. An attempt will be made to synthesize clinoptilolite from sodium aluminosilicate gels with Al:Si ratios of 1:3 and 1:6. Seeding with natural clinoptilolite will be tried, as well as the use of aluminosilicate glass. Preliminary results have shown a marked influence of the ionic environment on the ease of crystallization and nature of the products. A series of hydrothermal treatments of sodium aluminum silicates has been carried out in a synthesis program of the silicates. Ratios of Al:Si used were 1:6, 1:5, 1:4, 1:2, and 1:1. For temperatures of 250 to 350 C analcite and mordenite have been prepared. Their Al:Si ratios are 1:2 and 1:6, respectively. At higher temperatures albite and quartz result in the ratio range of 1:3 to 1:6. Two zeolites have been prepared from metakaolin and NaOH at 100 degrees C and these are the Linde zeolites A and X. A lithium zeolite was desired and it was synthesized from lithium hydroxide, metakaolin (Al2O3.2SiO2) and water at 90 degrees C. For extrudability attapulgite was added. In further experiments a new phase was produced in the high silica region which appears to be a layer-type mineral. Factors influencing the kinetics of the exchange reaction are crystal structure of the zeolite, size of the crystals and porosity of the particle. More investigations on these systems are planned. (NDV)

ADSORPTION; ALBITE; COLLOIDS; HYDROXIDES; ION EXCHANGE; LABORATORY STUDIES; ALUMINUM; SILICATES; SILICON; TEMPERATURE; ZEOLITES; MINERALS; ATTAPULGITE; ANALCITE; QUARTZ; SOLIDIFICATION; ENCAPSULATION

<607>
Salaszovich, S., H. Lauer, E. Merz, and D. Thiele, Institut fur Chemische Technologie, Kernforschungsanlage Julich GmbH, Munich, German Federal Republic.

FIPS - A Process for the Solidification of Fission Product Solutions Using a Drum Drier.

CONF-760310; STI/PUB/433; IAEA-SM-207/19; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 1, (pp. 315-323), 397 pp.(1976, March)

A new process, FIPS, has been developed for the solidification of fission product solutions from reprocessed HTGR fuel elements. The FIPS process consists of the following steps: concentration of fission product solution; denitration of the solution with formaldehyde; addition of glass-forming materials; drying of the slurry in a drum drier; melting of the dry product in a graphite crucible by rising level in-pot melting; and treatment of the off-gas and recovery of nitric acid. FIPS has been tested on a 1 kg glass/hr scale in hot cells using LWR fuel reprocessing fission product solutions since December 1974 without serious problems. Typical results of test runs were as follows: specific activity of waste 800 Ci/l; specific activity of the condensate from denitrification 5 x 10(E-2) to 2 x 10(E-1) Ci/l, decontamination factor 8000-2000; specific activity of slurry 2000 Ci/l; specific activity of condensate from drier 1 x 10(E-2) to 2 x 10(E-2) Ci/l, decontamination factor 2 x 10(E+5) to 1 x 10(E+5); specific activity of the glass 4-6 Ci/l. The glass matrix is 38% SiO2, 22% B2O3, 15% CaO, and 5% NaO. Approximately 20% fission product oxides are added. The melting point is 1150-1200 degrees C, with no "yellow phase" being observed. Most of the molybdenum is fixed as CaMoO4. Ru volatilization was 1% or less; off-gas analysis shows no selective Ru volatilization. Leach rates for the product range from 6 x 10(E-5) g cm(E-2)d(E-1) near the center of the block to 3 x 10(E-5) g cm(E-2)d(E-1) at the crucible-glass interface. These values are considered comparable to those of bottle glass. The high durability and simple elements of the process should make replacement of parts

<607> CONT.
relatively easy and overall costs low. A pilot plant with an output of about 10 kg glass/hr is under design. (LKB)

Total Ion Concentration

CONCENTRATION; DRUMS; DRYING; EQUIPMENT; FISSION PRODUCTS; GLASS; HOT CELLS; LABORATORY STUDIES; METHODS; NITROGEN COMPOUNDS; TRITIUM; THORIUM; ZIRCON COMPOUNDS; SILICATES; SEPARATION PROCESSES; SOLIDIFICATION; THERMODYNAMICS; WASTE TREATMENT; LEACHING; FUEL REPROCESSING; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, SOLID

<608>
Ball, A.P., J.T. Dalt J. Hudson, and J.A.C. Marples, United Kingdom Atomic Energy Authority, Atomic Energy Research Establishment, Harwell, England.

Development and Radiation Stability of Glasses for Highly Radioactive Wastes.

CONF-760310; STI/PUB/433; IAEA-SM-207/20; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 2, (pp. 3-14), 428 pp. (1976)

Various glass compositions were studied with regard to use in the vitrification of Magnox fuel wastes. Effects of variations in formation temperature, crystallization behavior, and leaching rate were recorded. Viscosities were measured between 800 and 1050 degrees C. The most important crystal phases were identified as magnesium silicate, magnesium borate, and ceria. Studies of stored energy release showed that energy is released over a wide range of temperatures, so that catastrophic triggering is highly unlikely. Helium release obeys Fick's diffusion law up to at least 30% of total inventory, with diffusion coefficients typical for borosilicate glasses. Leaching experiments using solid discs or slabs of glass in a Soxhlet apparatus at 100 degrees C revealed that the glass surface becomes depleted in Li, Na, and B, and that this layer dries to a granular powder which may become detached if thick enough. This depleted surface layer has been included in the total weight loss. Thus, tests on unpowdered glass samples will show leaching rates that decrease with time, whereas more massive samples will give almost constant rates. In testing alpha and beta resistance under exposure equivalent to 100 years, leaching rates for Pu-bearing glasses stored at 50 and 170 degrees C remained constant at approximately 1.5 x 10(E-2) g cm(E-2)d(E-1) after 10 days of leaching. Beta exposures in a Van de Graaff accelerator produced no observable changes. Exposure under an electron microscope had no effects at room temperature, but induced tiny bubbles in Li-glasses at 200 degrees C. Glasses without Li were radiation resistant, even at high temperatures. (LKB)

SOLIDIFICATION; PHYSICAL PROPERTIES; ALKALINITY; ZIRCON COMPOUNDS; CHROMIUM COMPOUNDS; GLASS; HELIUM; LABORATORY STUDIES; LEACHING; MOLYBDENUM; PLUTONIUM; PLUTONIUM COMPOUNDS; SILICATES; SODIUM COMPOUNDS; SULFUR COMPOUNDS; TEMPERATURE; WASTE MANAGEMENT; WASTES, HIGH-LEVEL; WASTES, SOLID; WASTES, RADIOACTIVE; IRRADIATION; LITHIUM

<609>
Hatch, L.P., and W.H. Regan, Jr., Brookhaven National Laboratory, Upton, Long Island, NY; American Water Works Association.

Processing of High-Level Atomic Wastes with the View to Ultimate Disposal.

TID-7517 (Part 1); Sanitary Engineering Aspects of the Atomic Energy Industry, Proceedings of a Seminar, Cincinnati, OH, December 6-9, 1955, (p. 184). (1956, October)

High level fission product waste solutions taken directly from the processing of spent uranium fuel elements from atomic reactors usually are not suitable for efficient adsorption of fission products by ion exchange on clay minerals. Separation of the bulk of the non-radioactive ions from the fission products, therefore, is a highly desirable first step. The use of such processes as electrodeionization, high temperature calcination, and distillation to accomplish such separations is discussed with emphasis on the elements of engineering design. (auth) (Complete Text)

SPENT FUELS; REACTORS; WASTES, HIGH-LEVEL; ION EXCHANGE; CLAYS; FISSION PRODUCTS; SEPARATION PROCESSES; CALCINATION; DISTILLATION; DESIGN; METHODS; WASTE TREATMENT; WASTE DISPOSAL; THEORETICAL STUDIES

<610>
Mebel, L.C., U.S. Congress, House of Representatives, Committee on Interior and Insular Affairs.

Nuclear Waste Management--Review of Commercial Nuclear Waste Management.

Oversight Hearings before the Subcommittee on Energy and the Environment, 95th Congress, First Session, Washington, D.C., May 16-17 and 20, 1977, Serial No. 95-15, (pp. 170-180), 315 pp. (1977)

The sources and types of nuclear wastes, management considerations and technical solutions, and performance goals and risk assessment are briefly discussed. From a light water reactor fissile uranium and plutonium results along with other transuranic (TRU) elements and fission products. Such wastes may be handled in one of three ways: disposal of the spent fuel; temporary storage of the fuel for later reprocessing; or reprocess and obtain the unused fissile uranium and plutonium. To date, reprocessing has been eliminated as an alternative due to the possibility of weapons proliferation. In addition to spent fuel other waste sources are: low-level waste contaminated with either fission products only or with TRU waste; high-level reprocessed wastes; uranium mine tailings; and gaseous effluents. High-level wastes are calcined and placed into canisters for storage in designated repositories. Spent fuel alternatives range from air cooled surface facility to geologic storage. The technology has been demonstrated for ten years. Geologic isolation appears at the present time to be more viable than deep space, ice sheet, or sea bed disposal. The U.S. progress on geologic disposal and the

<610>

WASTE PRODUCTION

<610> CONT.
need to understand the geology and hydrology of the formation are stressed. Low-level non-TRU wastes are buried in shallow trenches. A study has shown that tighter and more consistent regulation and institutional procedures are needed. Development of effective site criteria was deemed most important for successful site selection. Uranium tailings can be a long-term source of radioactivity to both the atmosphere and to the watershed in the region. Hence, the pile needs to be stabilized so future generations be endangered. At present radiocarbon, radiokrypton and radioiodine can be collected as gases; new technology will be required to gather radiotritium. There is controversy as to the basis of assessing risks: per capita, individual, or total population. The problem can be resolved if decisions can be made as to which risks are acceptable, various options for risk management, and the value of future technical implications. The wastes can be managed safely with existing technologies or extensions. (NDH) (RAP)

CONTAINMENT; CONTAMINATION; COST BENEFIT ANALYSIS; DOSE RATE; ENVIRONMENT; FISSION PRODUCTS; FUEL CYCLES; GEOLOGY; HAZARDS; HYDROLOGY; LEGISLATION; NUCLEAR POWER; HALF-LIFE, RADIOLOGICAL; REVIEWS; STANDARDS, FEDERAL; TAILINGS; WASTE MANAGEMENT; WASTE STORAGE; WASTES, GASEOUS; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, SOLID; WASTES, TRANSURANIC

<611>
Heiserl, W., Deutche Gesellschaft fur Wiederaufarbeitung von Kernbrennstoffen GmbH, Hcl, Belgium.

Techniques for High-Level Waste Solidification in Europe.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 21-29), 563 pp. (1979)

In some European countries considerable amounts of high-level waste solutions from nuclear fuel reprocessing already do exist or are expected to be generated within the next one or two decades. Among these countries are Belgium, France, Germany (Fed. Rep.) and Great Britian. It is commonly agreed that for a safe final disposal the liquid high-level waste has to be solidified. Therefore strong efforts are made to develop and demonstrate at industrial scale a process suitable for solidification. This paper gives a survey of the corresponding developments in the four European countries mentioned above in the past and present as well as the future plans. The HLW-solidification process under development (mainly AVH resp. AVH, HARVEST and PAMELA) are described and the characteristic features are shown. The processes and their products are compared. (Auth) (Complete text)

SOLIDIFICATION; WASTE TREATMENT; METHODS; WASTES, HIGH-LEVEL; FUEL REPROCESSING; REVIEWS

<612>
Hild, W., W. Kluger, and H. Krause, Gesellschaft fur Kernforschung mbH, Karlsruhe, German Federal Republic.

Bituminization of Radioactive Wastes at the Nuclear Research Center Karlsruhe - Experience from Plant Operation and Development Work.

KFK-2328; 12 pp. (1976, May)

This report summarizes some of the essential experience and results gained in radwaste bituminization at the Nuclear Research Center Karlsruhe both in plant operation and R and D activities. Due to very restrictive activity discharge limits, efficient waste treatment techniques had to be selected for the center; evaporation of low and medium level waste streams turned out to be the most effective and most economic procedure and was therefore adopted for a large treatment plant which started operation in 1968. The evaporator concentrates from that plant were solidified during the first years by mixing with concrete. This batching technique was replaced by the continuous bituminization process in 1972. The plant, the essential part of which is a self-cleaning screw-extruder evaporator, allows at the same time the effective evaporation of water from the evaporator concentrates and the homogeneous incorporation of the radioactive salt residues into bitumen. Bituminization must be limited to integral total doses of 10(E+8)-10(E+9) rads, corresponding to medium level wastes. In this connection, and due to the experience gained, there is no doubt that bituminization is very well suited for the solidification of waste concentrates from power reactors and will thus find a broad application in the future. As far as medium level waste streams from reprocessing plants are concerned, work is continuing to further increase the safety of this process by reducing both the activity and the nitrate content of the wastes. Optimum evaporation capacity is about 140 kg/H20/hr; the bitumen product is extremely homogeneous, with approximately 50 wt% salt and no more than 0.5 wt% residual water. The pH of the evaporator concentrate is adjusted to 8-10. Since start up in 1972, about 700 cu m of concentrates have been produced from over 40,000 cu m of low-and intermediate-level effluents containing 10(E+4) Ci. Approximately 2000 drums were produced with an average specific activity of 100 Ci/cu m and were shipped to the Asse salt mine for disposal. Distribution of dose rates was as follows: roughly 50%, 30%, 10%, and 5% of the drums had dose rates of up to 20, 40, 60, and 80 R/hr, respectively; the remainder showed dose rates up to 200 R/hr. To cope with this the sheet metal drums are either placed into a 200 liter reinforced drum that is inserted in special reusable shielded transport containers or the sheet metal drums are inserted into prefabricated concrete containers which after being topped off with concrete are transported to the Asse for disposal. (Auth) (LKH)

BITUMINIZATION; PLANTS, WASTE TREATMENT; WASTE TREATMENT; BITUMENS; EVAPORATOR CONCENTRATE; EVAPORATORS; SALTS; WASTES, INTERMEDIATE-LEVEL; WASTES, LOW-LEVEL; NITRATES; pH; CONTAINERS; DRUMS; METHODS

<613>
Hill, J., United Kingdom Atomic Energy

<613> CONT.
Authority.

Radioactive Waste Management in the United Kingdom.

Atom 240:250-253; CONF-760701; Management of Waste from the LWR Fuel Cycle, Proceedings of an International Symposium, Denver, CO, July 11-16, 1976, (8 pp.).(1976, October)

Processing of irradiated fuel at the Windscale plant was started some 25 years ago as part of the national defense program. Now, wastes from the nuclear power program, which is 20 years old, are being managed. At present, 5000 MW are generated by nuclear power and another 5000 MW are expected to be generated in the next three years. In the years the waste management program has been operating 20 tons of plutonium have been processed. The principles used in the United Kingdom for control of the nuclear industry are based on international recommendations. They are: 1) to ensure that no member of the public is exposed to radiation exceeding the dose limit; 2) to ensure that the whole population does not receive a dose more than 1 rem/person in 30 years; and 3) to do what is reasonably practicable to reduce dosen far below these levels. These objectives can be converted into a practical program as follows: 1) all unnecessary exposure to radiation should be avoided; 2) all unnecessary wastes should be avoided; 3) all unnecessary operations should be avoided and 4) all wastes must be put into a physical form which can be recovered. Wastes are handled in accordance with the above stated program. Gaseous wastes are treated according to the problem with high efficiency filtration, inertia desisting, electrostatic precipitation, or scrubbing, or absorption processes. Low active solid wastes almos free of contamination are buried in approved areas, while the other wastes contaminated with low beta/gamma activity and alpha activity are treated and disposed of in the ocean. High active solid wastes and fuel element cladding is stored. Sea disposal is used for low active liquid wastes after processing by distillation and floc precipitation. Medium active liquid wastes are treated by two methods. The first is to concentrate by evaporation, store for a specific time interval, and then discharge to the environment. An alternative is to treat by floc precipitation, treat the liquid as low level waste and dewater the sludge, and store. Storage in thick-walled concrete cells is the method used to handle highly active liquid waste. By the year 2000 it is expected that ten times the level of fission products will have to be handled. At present no major changes in waste management policy are expected, but with further research better procedures will be used. Also work is proceeding on vitrification. (NDV) (RAF)

ABSORPTION; ALPHA PARTICLES; BURIAL; CONTAINMENT; DECONTAMINATION; EVAPORATION; FISSION PRODUCTS; FLOCCULATION; RADIATION, GAMMA; PACKAGING; PRECIPITATION, CHEMICAL; REVIEWS; SLUDGES; STANDARDS, INTERNATIONAL; WASTE MANAGEMENT; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, SOLID

<618>
Jardine, L.J., and W.J. Steindler, Argonne National Laboratory, Chemical Engineering Division, Argonne, IL.

Metal Encapsulation of Ceramic Nuclear.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 181-189), 563 pp.(1979)

A discussion is presented on the merits of metal encapsulation. The process may be considered as a back-up to vitrification for all high-level wastes, including light water reactor wastes, or as the major solidification method for the fast breeder reactor wastes. Some of the attributes are: a decreased probability of dispersal by hydraulic or aeolian means as a result of mechanical damage; and an increased heat transfer with all more high-level wastes to be incorporated. A comparison with vitrification is made and problem areas are discussed. (NDV)

VITRIFICATION; ENCAPSULATION; SOLIDIFICATION; METAL METHODS; WASTES, HIGH-LEVEL

<615>
Jennings, A.S., Savannah River Laboratory, Aiken, SC.

Plan for Solidification of Savannah River Plant High-Level Waste.

DP-MS-77-23; American Institute of Chemical Engineers 70th Annual Meeting, New York, NY, November 13-17, 1977, (32 pp.).(1977)

Plans for the long-term management of high-level wastes at Savannah River Plant, presently stored as alkaline salts and sludges in underground tanks, are for their conversion to high-integrity solids for transport to a Federal Repository. Estimates are that by 1985, when waste removal is planned to begin, there will be 31 waste storage tanks containing 13.3 million gal of damp salts, 3.8 million gal of sludge, and 5.6 million gal of liquid waste. Hydraulic removal of salts from the tanks will require 42 million gal of water. This solution will be combined with supernate in the sludge storage tanks and circulated by in-tank pumps to slurry the sludge, producing 60 million gal of waste. This waste will be approximately 2.2 molar $NaNO_3$ and 1.1 molar $NaNO_2$ and should contain (by 1985) about 2.1 Ci/gal Sr 90, 2.2 Ci/gal Cs 137, 0.77 Ci/gal Pm 147, and 0.19 Ci/gal Ce 144-Pr 144. The sludge will be centrifuged out, taking with it most of the activity except for Cs 137; the remaining solution will be decontaminated by ion exchange, the salt evaporated to a solid and stored at SRP. The ion exchange eluate and sludge are combined, dried to a powder by spray calcination, combined with glass frit, and melted in a continuous electric melter. The melt is poured into stainless steel containers, solidified, the containers sealed, leak-checked, decontaminated, and placed in stainless steel overpacks for shipment. Off-gas from the melter/calciner is scrubbed, filtered, and released. (LKN)

SALTS; SLUDGES; WASTES, HIGH-LEVEL; WASTE MANAGEMENT; WASTES, LIQUID; ION EXCHANGE; EVAPORATION; CALCINATION; VITRIFICATION; SOLIDIFICATION; NITRATES; NITRITES; STRONTIUM;

<615>

WASTE PRODUCTION

<615> CONT.
CESION; WASTE TREATMENT; WASTE VOLUME

<616>
Johnson, A.J., and L.Q. Fong, Rockwell International, Atomics International Division, Energy Systems Group, Rocky Flats Plant, Golden, CO.

Waste Incineration and Immobilization for Nuclear Facilities (April-September 1977).

RFP-2783; 10 pp. (1978, October 20)

A fluidized bed incinerator using various pretreatment and feeding techniques has been developed to burn organic ion exchange resins with up to 55% water content. A reduction factor of 8:1 by volume, 7:1 by weight has been obtained by feeding the resin below the fluidized bed surface. Various methods of feeding the waste into the incinerator as an alternative to the present compression screw were investigated without success; an extrusion ram was found to suffer extensive damage from hard particles in the test waste. A bench-scale metallic vitrification chamber has been operated and has produced glass from incinerator residues and other waste materials. After extensive testing, Inconel 690 was found to be the best corrosion-resistant material for construction of the chamber. Use of glow bars instead of coiled wire resistance elements should significantly increase on-stream time. Two methods of vitrifying HEPA filters were used: (1) incineration of the entire filter and vitrification of the residue, and (2) separation of the filter media and the frame, incinerating the frame and vitrifying the media directly. The composition of HEPA filter components is given. (LKH)

INCINERATION; IMMOBILIZATION; WASTES, SOLID; RESINS; FILTERS; GLASS; RESIDUES; VOLUME REDUCTION; EQUIPMENT; METHODS

<617>
Johnson, J.B., S.D. Falkerson, T.W. McVay, and J.H. Warde, Oak Ridge National Laboratory, Oak Ridge, TN.

Heat Experiments for Self Fixation of Radioactive Wastes in Clay-Flux Mixtures.

TID-7517 (Part 1B); Sanitary Engineering Aspects of the Atomic Energy Industry, Proceedings of a Seminar, Cincinnati, OH, December 6-9, 1955, (pp. 339-351). (1956, October)

A method of disposal of high-level wastes would allow the heat of radioactive decay energy to fix a clay-flux waste mixture. This requires wastes containing an energy generation of at least several watts/gallon and is predicted on the development of workable clay-flux-mixtures which will adequately "fix" the wastes. The pits to be used would be at least 20 ft in diameter and 20 ft deep, lined with concrete and insulated with granular vermiculite. Several heat experiments were devised to determine the temperatures which well insulated dry clay masses would reach as a function of their surface to volume ratios for various specific energies. The area/volume ratios of 4:2:1 were used. It was found that a 20 foot diameter pit with a dry clay mass about 6 ft deep generating 0.005 watts/cc (1 Ci/cc) would reach 900 C. A second set of heat experiments was done to observe the frothing, boiling, fuming, drying, and sintering of the clay mixtures. The first run produced a final cake which was not stable to attack by water. In the second experiment the mixture became a frothy gel and filled the container. Heat was then introduced at a power level of 20 watts and the gel became a liquid and the level dropped gradually until after 40 hours at which time it was reduced to half the volume. The experiment was ultimately run at full power, 200 watts. The maximum equilibrium temperature was 850 C while the maximum average temperature was 780 C with a specific energy of 0.02 watts/cc. The final cake was found to be solid, free of cracks and well sintered. A pilot pit experiment is planned. While this method seems to be a good possibility, many problems remain, including evaluating the degree of volatility of fission products at high temperatures, the handling of waste from their source to the pits, and the operation of large numbers of such disposal pits. (NEW)

CLAYS; DECONTAMINATION; EVAPORATION; FISSION PRODUCTS; LABORATORY STUDIES; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, RADIOACTIVE; PITS

<618>
Johnson, L.F., British Nuclear Fuels, Ltd., Windscale Works, Seascale, England.

Operational Techniques to Minimize, Segregate, and Account for Alpha-Bearing Wastes.

CONF-760310; STI/PUB/433; IAEA-SM-207/86; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976, International Atomic Energy Agency, Vienna, Vol. 2, (pp. 191-200), 428 pp. (1976)

This paper summarizes procedures adopted at the Windscale Works of British Nuclear Fuels, Ltd. to reduce the volume of Pu-contaminated waste produced. Wastes are categorized as to disposal, treatment and storage routes available. For drummed Pu-contaminated waste, routes include sea disposal, incineration and interim storage pending development of further treatment methods. Methods presently available, and their limitations, are described. New equipment expected to become available in 1976, including a segmented gamma scanner, an NNC "random driver" neutron interrogation device, and a means of determining the Pu-content of crated waste, are described. Also described is the data management system, which provides information on the amounts of Pu lost to waste and which is used to account for the fissile material in each area. Figures used in accounting are calculated from non-destructive analyses at the waste handling facility. As a result of these techniques, the weight of waste arising annually from 1973 through 1975 has decreased progressively by 36% and the weight of associated Pu by 57%. (LKH)

EQUIPMENT; INSTRUMENTS; WASTE MANAGEMENT; WASTE VOLUME; WASTES, RADIOACTIVE; RADIOACTIVITY; METHODS; VOLUME REDUCTION; PLUTONIUM

<619>
Kanno, T., and H. Himura, Research Institute of Mineral Dressing and Metallurgy, Tohoku University, Sendai, Japan.

Processing of Radioactive Waste Solution with Zeolite (2). Leachabilities of Cesium and Strontium from Calcined Zeolites.

Nihon Genshiryoku Gakkaishi 19(2):83-88; OLS-78-315.(1977, February; 1978)

Various types of zeolites calcined at high temperatures were investigated for leaching rates. Type A zeolite had an ion exchange capacity for Cs 137 of 178 meq/100 g, while types X and Y zeolites and mordenite had capacities of 185, 185 and 196 meq/100 g, respectively. For Sr 85 the ion exchange capacities were 590, 870, 290, and 210 meq/100 g. Type A shows the least capacity because only 4 of the Na's in the unit cell which contains 12 are exchanged with Cs. Apparently only those Na's existing near the eight membered-ring are exchanged with Cs. In the case of mordenite, out of 8 Na's in a unit cell 4 seem to exist in a side channel formed by the eight member-ring, and the rest in the largest side channel formed by the 12 membered-ring. This explains the high ion exchange capacity. When the calcining temperature is raised, leachability decreases as a result of the formation of pollucite at temperatures between 900 and 1000 degrees C. Cesium leaching rates for distilled water ranged from 2.5 x 10(E-8) to 7.8 x 10(E-8) g/sq cm d for the zeolites and mordenite. Strontium leaching rates were less than 2.6 x 10(E-10) to 5.0 x 10(E-9) g/sq cm d. Leaching rates rapidly decreased as time elapsed. Mordenite, if it is formed into a calcined block at high temperatures will have a low leaching rate, and become usable. Japan has large deposits of mordenite. (NDV)

CALCINE; CESIUM; ION EXCHANGE CAPACITY; LEACHING; MORDENITE; POLLUCITE; STRONTIUM; ZEOLITES; SOLIDIFICATION; WASTE TREATMENT; TEMPERATURE; WASTES, HIGH-LEVEL; LABORATORY STUDIES

<620>
Kelley, J.A., and W.H. Rankin, Savannah River Laboratory, Aiken, SC.

Correlation of Radionuclide Leachabilities with Microstructures of Glasses Containing Savannah River Plant Waste.

DP-1411; 15 pp.(1976, May)

Leachabilities of Cs and Sr from the glass matrices were correlated with the microstructures of glasses containing waste sludge from two Savannah River Plant waste tanks. All sludge components, except Hg which volatilized, were completely soluble in the melt, but small amounts of crystalline iron oxide precipitated during cooling. These precipitates have no effects; their leachabilities are less than those of their glass matrices and they are not enriched in Cs and Sr. The glass containing 40 wt% Tank 13 sludge devitrified when heated at 600 C for 1 month, as shown by DTA, X-ray, and microscopic analyses. Devitrification increased the Cs leachability by a factor of approximately 100 but it still was very low [< 10(E-6) g/sq cm day]. The glass containing 40 wt% Tank 5 sludge did not devitrify with the same heat treatment even though the two sludges have similar chemical compositions. (Auth) (NAP)

Leaching Rate

Sr 90; Ce 144; Ru 106; Eu 154; Cs 137; Sb 125

LEACHING; VITRIFICATION; DRY PRIFICATION; CESIUM 137; CHEMICAL COMPOSITION; WASTES, HIGH-LEVEL; STRONTIUM 90; CERIUM 144; RUTHENIUM 106; EUROPIUM 154; ANTIMONY 125; GLASS; SLUDGES; MERCURY; VOLATILITY; IRON; OXIDES; THERMAL PROPERTIES; PRECIPITATION, CHEMICAL

<621>
Kibbey, A.H., and H.W. Godbee, Oak Ridge National Laboratory, Oak Ridge, TN.

Physicochemical Characterization of Solidification Agents used and Products Formed with Radioactive Wastes at LWR Nuclear Power Plants.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, C.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 895-898), 563 pp.(1979)

The radioactive wastes routinely generated onsite at light-water-cooled nuclear power plants (LWR) comprise the largest volume of radwastes generated annually in the commercial nuclear fuel cycle (not including mill tailings and spent fuel). The principal wastes considered here are the evaporator concentrates, filter sludges, and spent ion exchange resins that are derived from cleanup of the liquid streams recirculating within or being discharged from LWR plants. Until now, most filter sludges and spent resins have been dewatered but were not incorporated into a solid matrix, as will be required in the future. The introduction of solidification agents to immobilize these sludges and resins can increase the volume of these wastes by a factor of one to two. The agents and methods used or proposed for use in solidification of LWR power plant wastes are generally suitable for treating most of the other-than-high-level wastes generated throughout the entire fuel cycle. The four solidification agents under consideration are cements, urea-formaldehyde (UF) resins, asphalts, and unsaturated polyester resin. Portland cements were the principal solidification agents used in the U.S. until 5 years ago when UF resins emerged. Advantages of UF resins included an all-liquid system to the point of solidificsiton, and lack of dusting and flash or premature setting problems of cements. However, frequent observation of unbound liquid in the "solidified" waste made it unsuitable for burial. Asphalts have been used in countries other than the U.S. Unsaturated polyester resins are a recent development as solidification agents, and have not yet been used routinely. Factors to shelf-life, special storage requirements of handling problems; flexibility regarding waste types, activity levels, and product containers; compatibility with wastes; and product properties such as leachability, density, dry waste solids content, product/liquid waste volume ratio, radiation resistance, mechanical resistance, and maximum resistance, and maximum stable temperature. More research is needed to aid future decision making in handling nuclear

<621>

WASTE PRODUCTION

<621> CONT.
wastes. (Auth) (DCW)

Two figures show annual thermal output and annual waste volume from LWRs in the U.S. Two tables give waste solidification agents used or proposed for use at LWRs, and considerations that influence the choice of solidification agents.

Waste Volume

WASTES, COMMERCIAL; WASTES, RADIOACTIVE; WASTE STORAGE; WASTE DISPOSAL; WASTE PROCESSING; IMMOBILIZATION; SOLIDIFICATION; CEMENTS; ASPHALT; RESINS; UREA FORMALDEHYDE; REACTORS, LIGHT WATER

<622>
Klingler, L.W., D.H. Batchelder, and R.L. Lewis, Hound Facility, Miamisburg, OH.

TRU Waste Cyclone Drum Incinerator and Treatment System: July-September, 1976.

MLM-2575; 22 pp.(1976, November 30)

Demonstration of the cyclone incinerator was completed, with all less than 10 nCi/g waste generated at the Mound Facility since December 1976 having been burned. Total weight for all waste burned was over 9 tons (1735 cu ft). Various steel alloys were tested for corrosion rate under conditions similar to those in the most corrosive area of the incinerator. Comparison of relative corrosion rate with materials and fabrication costs indicated that Hastelloy X is by far the best alloy tested (relative performance rating 24.3, where SS 304L = 1). Various types of dryers are being tested for the evaporation of aqueous byproduct streams from the incinerator (mainly the wet-scrubber offgas system). Final selection of equipment will be made after testing of the rotary vacuum dryer. Upgrading of the incinerator system was continued with the design and installation of a new lid assembly to replace a secondary chamber above the incinerator; other improvements include a new non-pneumatic ash-handling system, a drum movement and lifting system, pH control system for incinerator scrubber liquor, and improvements in the incinerator feed system. Waste immobilization studies included determination of ideal bone char/cement pellet matrix; a 70/30 ratio was determined to be best based on crush strength values. The Quality Control Program discovered that all previous crush strength data were in error; a correction graph is provided. Thirteen batches of 100 pellets have been pressed from Pu 238-contaminated sintered incinerator ash with less than 0.1% carbon, and an ash/cement ratio of 65/35. Pellets were tested for crush strength and thermal effects. Approximately 70% of the pellets tested showed visible damage under 3700 psi pressure. After heating to 750 C with no visible effects, another set of pellets was found to have crush strengths 25% lower than unheated pellets. Sludge/cement pellets of 60, 65, and 70% sludge had crush strengths generally below 2100 psi; work is planned on lower percentages of sludge to achieve acceptable strengths (around 3500 psi). Testing procedures for sludge/cement pellets are being developed. The new glove box line for immobilizing ash produced from incineration of less than 10 nCi Pu 238/g waste was completed. (Auth) (LKM)

INCINERATION; VOLUME REDUCTION; WASTES, LOW-LEVEL; ALLOYS; DRYING; ASHES; CEMENTS; CONCRETES; PLUTONIUM; THERMAL PROPERTIES; SLUDGES; GLOVE BOXES; IMMOBILIZATION; WASTE TREATMENT; WASTES, TRANSURANIC

<623>
Kennst, J.W., and R.R. Herald, Mound Facility, Miamisburg, OH.

Evaluation of New Macroporous Resins for the Removal of Uranium and Plutonium from Waste Streams.

MLM-2320; 8 pp.(1976, December 20)

Improved macroporous ion exchange resins were evaluated for removal of Pu 238 (+4), Pu 238 (+6), and U 233 (+6) from aqueous solution. Ion exchange capacities and distribution coefficients (Kd) were compared with each other and with bone char. It became apparent that pH values could not be maintained due to adsorption of the buffering agent and exchange for H+ and OH-; the various types of adsorbents demonstrated that the extent of removal and Kd's were a function of pH. Best removal of polymeric Pu 238 (+4) was achieved using bone char (hydroxyapatite) at pH 7 (Kd = 1.2 X 10(E+6)); however, Amberlite XE270, a weak base anion exchange resin, also showed reasonable Kd values (1.56 X 10(E+4)) at pH 7. For removal of ionic U 233 (+6), the strongly acidic cation exchangers gave the best results, mainly at pH's of 4-7 (Amberlyst XN1010, 9.5 X 10(E+6) at pH 7; Amberlite 200, 1.05 X 10(E+6) at pH 7); for anionic resins the better results were at pH 7-10. Gel resins performed poorly compared with the macroporous resins; both cationic and anionic macroporous resins gave Kd values on the order of 10(E+5) at various pH's, while gel resins gave Kd's on the order of 10(E+2). Best removal of Pu 238 (+6) was by Amberlyst XN1010, a strong acid macroporous cation exchange resin (Kd = 6.23 X 10(E+5) at pH 7); the gel resin Dowex-SER-P gave lower Kd values than bone char. As this study was only a preliminary screening, no concern was shown for the fact that pH changed as solute removal was taking place. The capacity used to determine Kd values was not the theoretical total capacity of the material; however, it was felt that for purposes of comparison, Kd's would not change significantly with further loading. Additional tests are planned to determine isotherms for each adsorbent material, loading rates, bed capacity, residence time and removals for the various resins, and if it is possible to maintain constant pH. (LKM)

Ion Exchange Capacity; Distribution Coefficient

Pu 238; U 233

RESINS; ION EXCHANGE; EVALUATION; LABORATORY STUDIES; SORPTION; PLUTONIUM; URANIUM; WASTE TREATMENT; pH; SEPARATION PROCESSES; VALENCE

<624>
Kolychev, B.S., A.N. Matveev, and A.A. Khonikevich.

Handling of Radioactive Waste at Soviet Nuclear Power Stations.

<624> CONT.
Atomnaya Energiya 26(5):811-819; Soviet Atomic Energy 26(1):869-878.(1969, January; 1969, May)

The author's conclusions in reviewing the topic are as follows: The problem of dealing with all types of radioactive waste at Soviet Nuclear power plants has been solved, and therefore these plants can be sited in any part of the country. In order to decontaminate the primary cooling water, filters should be installed which have an alluvial layer, and ion exchange capacity. Evaporation plants can also be used, but their operation is more costly. Burial in deep geological formations is a promising and economical method for disposing of salt-containing liquid waste of medium specific activity. However, this method requires a suitable isolated high-capacity collector which should be taken into account when choosing a site for a nuclear power plant. Water from the fuel element cooling ponds should be decontaminated in special by-pass installations consisting of filters with an alluvial layer and ion exchange filters. Liquid waste and pulp from the plants for decontaminating waste water should be consolidated by cementation or bitumenization. Solid radioactive waste should be transformed into monolithic concrete masses, by filling up with cement mixed with liquid waste of low and medium activity. In order to decontaminate air it is recommended that two-stage filtration plants and gas holders be constructed and disposal is effected through gas-discharged stacks. (Auth) (NAP)

Diagrams show storage vaults for liquid and solid wastes, an evaporation plant for the decontamination of primary circuit water, decontamination steps using coagulation, evaporation, and ion exchange, pumping system of low-level liquid wastes into deep geologic formations, and a cementation plant for liquid wastes.

REVIEWS; NUCLEAR FACILITIES; WASTES, RADIOACTIVE; STORAGE, GEOLOGIC; WASTES, SOLID; WASTES, LIQUID; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; ION EXCHANGE; FILTERS; EVAPORATION; CONTAINERS; TRENCHES; COOLANTS; DECONTAMINATION; WATER; SITE SELECTION; COST BENEFIT ANALYSIS; BURIAL; CEMENTS; CONCRETES; BOREHOLES; GASES; KRYPTON 85; XENON 135; IODINE 131; STRONTIUM 90; YTTRIUM 90; CERIUM 144; MONITORING; WASTE TREATMENT; BITUMINIZATION; WASTE STORAGE; WASTE DISPOSAL

<625>
Krause, H., Karlsruhe Nuclear Research Center, Karlsruhe, German Federal Republic.

Studies on Treating Radioactive Wastes from Reprocessing Installations.

KFK-2255; OLS-78-285; First Status Report of the Project on Reprocessing and Waste Treatment, Collection of Lectures, November 17, 1975, (pp. 7/1-7/18).(1976, August; 1978)

Treatment of radwastes at reprocessing plants, particularly Karlsruhe is discussed. A total of 320,000 cu m of high-level radwaste is stored all over the world. Some of the older tanks in the U.S. have leaked but the newer ones made of high quality steel retain their integrity. Solidification products such as calcinates and glasses are used extensively for high-level wastes. The glasses have proved far superior since they only allow less than 1% of the mass to be leached in 500 years. Solidification processes, such as evaporation, denitration, drying, calcination and melting are briefly discussed. The medium-level radwastes are very important in terms of both volume and their fission product and actinide content. At Karlsruhe both the low- and medium-level wastes are concentrated together and fixed in bitumen. About 2,000 cu m have been processed in this manner. About 70,000 cu m of low-level radwastes have been evaporated at Karlsruhe. Organic solvents and fuel element cladding are also treated at Karlsruhe. The volume of burnable radwastes have been reduced by incineration and the ashes converted to a solid by incorporation with concrete. Alpha wastes from uranium ore, Oklo, and other sources along with final storage and decontamination are briefly discussed. While many processes are at the pilot plant stage much remains to be done to bring them to large scale technological maturity. (NDV)

Waste Volume

ACIDS, ORGANIC; ASHES; BITUMENS; CALCINATION; CLADDING; COMBUSTION; CONCRETES; DECONTAMINATION; DRYING; ENCAPSULATION; EVAPORATION; FISSION PRODUCTS; ACTINIDES; PLANTS, WASTE TREATMENT; DENITRATION; WASTE TREATMENT; VOLUME REDUCTION; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, SOLID; SOLIDIFICATION

<626>
Kul'bashnaya, N.V., and Ye.K. Fosin.

Influence of Surfactants on Bitumenization of Radioactive Waste.

Soviet Atomic Energy 22(1):496-498; Atomnaya Energiya 22(5):397-399.(1967, January; 1967, May)

Experiments were conducted on the viscosity effects of surfactants in the bitumenization of radioactive waste. For the purpose of these tests, dry inert residue obtained from iron hydroxide pulp by the "oil" flotation method was used. This residue contains 20-50 wt% surfactants adsorbed on the iron hydroxide, mostly petroleum sulfo acids and synthetic detergents. The binder was BN-3 bitumen (softening point 50 C). Using a mixture of 60% bitumen, 40% residue at a constant temperature of 185 C, thickening was observed with prolonged heating beginning at 34% surfactant in the residue (13.5% in the compound). With 70% surfactants in the residue (28% in the compound), viscosity increases very rapidly. Thickening apparently occurs by formation of solvate coatings around the mineral particles of the inerts. The solvate particles of the dispersed phase occupy larger volumes, and react with the bitumens and with one another to form a structurized system whose viscosity increases as the sol passes into a gel. An equation describing the relation of viscosity and temperature for compounds containing less than 60% inerts (27% surfactants) is given. This allows viscosity assessment to within plus or minus 10%, and determinations of stress on the stirrer and the diameter of the discharge pipe. Above 27% surfactants, viscosities increases too rapidly to permit calculation. From these experiments it was determined that the maximum permissible surfactant content for radioactive waste

<626>

WASTE PRODUCTION

<626> CONT.
bituminization should be 20% in the compound, permitting 180-190 C temperatures for 2-3 hrs without marked viscosity increases. (LKH)

BITUMINIZATION; BITUMENS; VISCOSITY; SURFACTANTS; DETERGENTS; EQUATIONS; LABORATORY STUDIES; TEMPERATURE; WASTES, RADIOACTIVE; IMMOBILIZATION; WASTE TREATMENT; THERMAL PROPERTIES

<627>
Kupfer, M.J., D.M. Strachan, and W.W. Schulz, Rockwell Hanford Operations, Richland, WA.

Technical Challenges in the Immobilization of Hanford Defense Waste.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 891-894), 563 pp. (1979)

Hanford high level radioactive defense waste is highly varied in chemical composition and poses a unique problem to the development of glass and ceramic waste forms for its immobilization. Chemical variability is the principal difference between Hanford defense waste and other nuclear wastes, including proposed commercial waste and other defense wastes. Upon completion of the current high-level waste solidification program, approximately 95000 cu m of salt cake, 53000 cu m of sludge, 35000 cu m of slurry, and 11000 cu m of complexed concentrate (residual liquid) will be stored in underground tanks. An additional four to six exabecquerels (110 to 160 megacuries) of CsCl and SrF2 will be doubly encapsulated and stored under water. Salt cake consists of water soluble sodium nitrate containing small amounts of Cs 137 and Sr 90. The evaporation process to produce salt cake leaves a highly alkaline slurry and complexed concentrate liquid containing ions of Na, OH, AlO2, Cs 137, TcO4, Sr 90. The Cs and Sr are being removed, doubly encapsulated, and stored as CsCl and SrF2. Sludge is the water insoluble mixture of hydrated metal oxides, hydroxides and phosphates that precipitate when highly acidic process wastes are made alkaline for storage. Sludges represent the most highly variable waste type because little blending of these wastes has occurred since their generation from the four separation processes: Bismuth Phosphate, Uranium Recovery, Redox, and Purex. Their principal radionuclide contents are Sr 90 and Pu 239. A leading option under consideration for the long term management of Hanford defense waste involves waste retrieval and dissolution of the salt cake and slurries in water. Long lived radionuclides would be removed from the liquid converted to an insoluble ceramic or vitreous waste form. Some laboratory studies for the fixation of these waters into glass and ceramic forms have been reported and are continuing. (Auth) (DCW)

One table is presented which gives typical Hanford sludge compositions.

Waste Volume

Sr 90; Cs 137; Pu 239; Tc 99

WASTES, DEFENSE; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTE STORAGE; WASTE DISPOSAL; GLASS; CERAMICS; IMMOBILIZATION; SOLIDIFICATION; ENCAPSULATION; CHEMICAL PROPERTIES; WASTE MANAGEMENT; WASTE PROCESSING

<629>
Lamb, K.B., and H.S. Cole, Idaho National Engineering Laboratory, Idaho Falls, ID.

Development of a Pelleted Waste Form for High-Level Idaho Chemical Processing Plant Zirconia Wastes.

ICP-1185; 32 pp. (1979, April)

The preparation of a pelleted waste form from calcined waste using a rotating disc pelletizer has been tested as a possible candidate process for long-term management of high-level radioactive defense wastes generated at the Idaho Chemical Processing Plant (ICPP). Simulated zirconia calcine was formed into a 3-10 mm diameter spherical pellets using a liquid binder of 7M H3PO4-4M HNO3 and a solid binder consisting of 5 wt% bentonite, 2 wt% metakaolin and 2 wt% boric acid. After heat treatment at 900 C for 2 hours, the pellets were impact resistant and had a leach resistance of 10(E-8)g/sq cm/day, based on Soxhlet leaching for 100 h at 95 C with distilled water. Extensive binder studies involving a total of 32 pellet composition with zirconia calcine were made. Liquid binders tested were phosphoric acid, nitric acid, silica, and alumina; solid binders tested were bentonite, metakaolin, boric acid, and lime. Pelletizing high-level zirconia calcine may be a practical alternative to vitrification of calcine storage. Pelletizing reduces the surface area of calinated waste by a factor of 1000 or more. An integrated pilot plant is being built based on this pellet development work. (NAF)

Leaching Rate; Temperature

WASTES, HIGH-LEVEL; PELLETS; ZIRCALOYS; CHEMICAL COMPOSITION; CALCINE; EQUIPMENT; BINDERS; BENTONITES; KAOLIN; BORIC ACID; PHOSPHORIC ACID; NITRIC ACID; LEACHING; CESIUM; SILICA; LIME; LABORATORY STUDIES; ALUMINUM COMPOUNDS

<629>
Larker, H.T., ASEA, Robertsfors, Sweden.

Hot Isostatic Pressing for the Consolidation and Containment of Radioactive Waste.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 207-210), 563 pp. (1979)

Hot isostatic pressing as a technique for containing radioactive waste is presented. The technique utilizes argon gas at between 100 and 300 MPa. Very good fixation, in some instances better than the best glass compositions, has been achieved as well as high waste loading. As an application of this method, burned out fuel rods were placed in a container of high density and highly pure alpha alumina (corundum) and treated under high temperature and pressure. More work is proceeding in an effort to achieve

<629> CONT.
full scale operations. (NDV)

ALUMINUM COMPOUNDS; CONTAINERS; ENCAPSULATION; METHODS; FUEL RODS; FIXATION; WASTE TREATMENT; LABORATORY STUDIES

<630>
Lawrence, W.C., Battelle-Pacific Northwest Laboratories, Richland, WA.

Preliminary Concept of Waste Retrieval System-Tank Wall Access Using Underground Mining Techniques for Removal of Radioactive Waste (Sludge, Saltcake and Liquid).

BNWL-2234; 23 pp. (1977, February)

A preliminary evaluation of conceptual mining systems to retrieve solidified high-level radioactive waste stored in the Hanford underground waste storage tanks was performed. Operation of the conceptual mining equipment will involve several steps: site preparation, equipment placement and containment precautions, and mining. The entire equipment package consisting of control trailer, hydraulic power supply, waste spade unit, TV camera, lighting and audio van, and pumps or other support equipment is mounted on special trailers built to handle load requirements. The remotely-controlled mining concept is designed to handle all forms of waste (solid objects, saltcake, sludge, liquid, and combinations thereof) from all sizes of waste tanks (500,000 to 1,000,000 gal) at a rate greater than 50 tons/day. It will be fully capable of remote operation, and would be equipped to clip or saw internal metal components for tank access or structural component removal as required. The system is designed to transport waste material directly from the subsurface storage tanks to a surface waste transporting process. (Auth) (NAP)

RETRIEVABILITY; WASTES, HIGH-LEVEL; WASTES, SOLID; WASTES, LIQUID; WASTE STORAGE; WASTE VOLUME; THEORETICAL STUDIES; TRANSPORTATION; EQUIPMENT; PITS; CONTAINMENT; SHIELDING; TANKS; SLUDGES

<631>
Lee, J.L., and J.W. Rhodes, Idaho National Engineering Laboratory, Idaho Falls, ID.

Leak Testing Fuel Stored in the ICPP Fuel Storage Basin.

ICP-1110; 11 p. (1977, June)

Irradiated fuel waiting to be reprocessed at Idaho Chemical Processing Plant (ICPP) is stored under water in the fuel receiving facility. By December 1976 the stored fuel was leaking fission products at a rate of about 80 Ci/wk. While ion-exchange columns maintained the total concentration at 0.1 uCi/ml, the radiation personnel were receiving excessive exposure. It was determined by isotopic ratios that the stainless steel-type EBR-II fuel was the major contributor. The equipment used for testing the subassemblies responsible for the leakage was the Fuel Element Leak Tester (FELT). A suspected subassembly was placed in the FELT and the vessel was purged with running water. About 25 complete FELT column changes of water were used. After the rinse period was completed, the water flow was stopped and a two hour test period began. The water was then removed, analyzed, and the fission product leak rate calculated. Any leak rates up to 0.05 Ci/day were attributed to desorption of fission products and considered non-leakers. Those subassemblies however, that leaked at rate greater than 10 times the lower limit were considered apparent major leakers. The highest leaker was taken for disassembly and examination. The fuel rods were damaged and hence highly susceptible to leaching. Possible causes of the damage include localized overheating in the reactor or the shipping cask or corrosion during storage in basin water. The leak test proved to be a valid test method for identifying fuel that is leaking significant quantities of radionuclides during water storage. (NDV)

CASKS; CONTAINER INTEGRITY; CORROSION; FISSION PRODUCTS; FUEL ELEMENTS; FUEL REPROCESSING; IRRADIATION; ISOTOPE RATIOS; LEAKS; MONITORING; RADIATION DETECTORS; WASTE STORAGE; WATER

<632>
Lewis, A.E., J.J. Cohen, and R.L. Braun, Lawrence Livermore Laboratory, Livermore, CA.

Disposal of Nuclear Waste by In-Situ Incorporation in Deep Molten Silicate Rock.

Memoir 18; Underground Waste Management and Environmental Implications, T.D. Cook (Ed.), Proceedings of a Symposium, Houston, TX, December 6-9, 1971. American Association of Petroleum Geologists, Tulsa, OK, (pp. 280-286). (1972)

A method for disposing of nuclear-reactor wastes by in-situ incorporation in molten silicate rock involves injecting liquid wastes into a deep underground chimney and allowing them to self-boil. The resulting steam is processed at the ground surface and recycled into a closed system. When waste addition is terminated, the chimney is allowed to boil dry, thereby solidifying the waste. The heat generated by the radioactive waste then melts the surrounding rock, which dissolves the waste. Finally, the rock resolidifies, trapping the radioactivity in an insoluble rock matrix deep underground. This method has significant environmental and economic advantages over previously proposed methods. Storage and processing are minimized and the need for transportation of the wastes is eliminated, thereby significantly reducing the potential for accidental releases. The cost of high-level waste disposal using state-of-the-art methods is estimated to be between $8,000 and $12,000 per ton of fuel processed. The method proposed here is estimated to cost about $2,000 per ton. (Auth) (NAP)

Artist's concept of waste disposal in a nuclear chimney and sequence of events is illustrated.

WASTES, LIQUID; STORAGE, GEOLOGIC; WASTE DISPOSAL; SOLIDIFICATION; SILICATES; EQUATIONS; THERMAL PROPERTIES; WASTES, HIGH-LEVEL; HEAT TRANSFER; ROCK MECHANICS; COST BENEFIT ANALYSIS; SITE SELECTION; THEORETICAL STUDIES; WELLS, INJECTION

<633>

WASTE PRODUCTION

<633>
Logsdon, J.E., and J.W.W. Hickey, U.S. Environmental Protection Agency, Washington, D.C.; Notre Dame University, South Bend, IN.

Radioactive Waste Discharges to the Environment from a Nuclear Fuel Reprocessing Plant.

Radiological Health Data and Reports 12(6):305-312 (1971, June)

Construction of the first commercial nuclear fuel reprocessing plant was completed in June 1965 and it was licensed in 1966. Its function is to separate plutonium and unused uranium from associated fission products in the spent fuel. The western New York Nuclear Service Center is a 3,331 acre preserve with the plant 1,200 meters from the nearest boundary. A fence surrounds the NFS plant, lagoons, waste tanks, and burial grounds. The plant is able to reprocess 1 metric ton/day of any low-enriched nuclear fuel which can be reduced to a nitric acid solution. A minimum of 150 days of cooling is required before processing and most fuel is cooled over a year. The fuel is delivered by rail or truck in shielded casks which are opened in a 40 ft deep water tank. The fuel elements are then placed in storage canisters. After an appropriate amount of time the canisters go to the Process Mechanical Cell where the elements are chopped into pieces, put into baskets and stored in the General Purpose Cell for later chemical processing. After chemical processing and purification the uranium and plutonium are packaged in geometrically safe containers for storage and shipment. The total operation takes 30 to 60 days and can involve up to 60 metric tons of spent fuel. The primary sources of gaseous radwaste are the chopping and dissolution processes. Krypton 85 and I 129 are the most important nuclides. Krypton usually decays within 8 days and insufficient data exists on I 129. The gas is put through scrubbers and filters before release. Low-level waste comes from the condensate of the high-level evaporators. It is discharged to the lagoons for 1 month and then put into a nearby creek. The high-level waste comes from the extraction processes and this waste is stored in steel tanks. Solid waste is buried on site in the silty till. Since the creeks are used for recreation, indirect exposure pathways exist and an appropriate monitoring program is in effect. It appears that the operation of the plant has not increased the doses of radionuclides. (NDV)

Waste Volume

BIOSPHERE; BURIAL; CLADDING; DECONTAMINATION; DOSE RATE; ENVIRONMENTAL EXPOSURE PATHWAY; EVAPORATORS; FILTERS; FUEL REPROCESSING; PLANTS, WASTE TREATMENT; SHIELDED CONTAINERS; TANKS; WASTES, GASEOUS; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTES, SOLID; WASTES, LOW-LEVEL

<634>
Lethy, D.V., and W.H. Bond, Mound Facility, Miamisburg, OH.

Volume Reduction System for Solid and Liquid TRU Waste from the Nuclear Fuel Cycle: October-December 1977.

MLM-2509; 11 pp. (1978, April 24)

The laboratory-scale cyclone incinerator at the Mound Facility was assembled and equipment check-out begun during the October-December 1977 reporting period. Modifications to a glove box to be used for the demonstration phase of incinerator ash immobilization were approximately 80% completed, with demonstration to begin next quarter. Pellets of bone char, ash, and sludge-ash in various mixtures were tested to establish pressing and curing parameters. Curing studies were conducted to define differences in dry and wet cured pellets. Leach tests were performed at room temperature, 70 degrees C, and 100 degrees C; leach rates at 70 degrees C were about 10% greater than at room temperature; at 100 degrees C the rate was 99% higher than at room temperature. Wet curing produced lower leach rates. Long-term leach tests on Pu 238-contaminated ash/cement pellets in water showed that the highest leach rates were from 60% ash mixtures. All rates were on the order of 10(E-11)-10(E-12) g/sq cm/hr. (Auth) (LKH)

Leaching Rate

Pu 238

INCINERATION; CEMENTS; ASHES; DESIGN; EQUIPMENT; LABORATORY STUDIES; SLUDGES; TRANSURANICS; WASTE TREATMENT; WASTES, RADIOACTIVE; WASTES, SOLID; WASTES, TRANSURANIC; PHYSICAL PROPERTIES; VOLUME REDUCTION; IMMOBILIZATION

<635>
Latze, W., J. Borchardt, and A.K. De, Hahn-Meitner-Institut für Kernforschung Berlin, GmbH, Berlin, German Federal Republic.

Characterization of Glass and Glass Ceramic Nuclear Waste Forms.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 69-81), 563 pp. (1979)

Since problems exist with solidifying high-level liquid wastes in both the borosilicate and phosphate glasses, glass ceramics were more closely investigated. Recent studies include: long term phase stability (cyprite/h-celsian/a-celsian), energy released during crystallization, and actinide nuclide distribution. Celsian type was found to be the most favorable material. A new type of glass ceramic, fresnoite ($Ba_2TiSi_2O_8$) has been developed. This glass ceramic is boron free and still homogeneously digests fission product oxides. It easily accomodates Sr in the host phase and has a decreased leachability in the residual glass phase due to a lack of boron. Glasses more stable than the former can be produced with melting temperatures of the parent glasses up to 1670 K if ceramic electrodes are available. Product characterization will be performed under the criteria of chemical, thermal, mechanical, and radiation stability. Comparison of the products show a high standard of quality. (NDV)

SOLIDIFICATION; ENCAPSULATION; GLASS; CERAMICS; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; TEMPERATURE; RADIATION EFFECTS; WASTES, HIGH-LEVEL; WASTES, LIQUID; LABORATORY STUDIES

<635>
Lynch, R.B., R.G. Couch, D.T. Ramna, J.K. Johnstone, and E.J. Bonk, Sandia Laboratories, Albuquerque, NM.

The Sandia Solidification Process - A Broad Range Aqueous Waste Solidification Method.

IAEA-SM-207/75; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, (pp. 361-378), 424 pp. (1976)

New ion exchange materials of the hydrous oxide type have been developed for solidifying aqueous radioactive wastes. These materials have the general formula $H(H^{+}xOyHz)n$, where H is an exchangeable cation of charge +n and H' may be Ti, Nb, Zr, or Ta. Affinities for polyvalent cations were found to be very high and ion exchange capacities large, e.g. 4.0-4.5 meq/g for $HaTi2O5H$, depending on moisture content. The effectiveness of the exchangers for solidifying high-level waste resulting from reprocessing of light water reactor fuel was demonstrated in small-scale tests. Used in conjunction with anion exchange resin, these materials reduced test solution radioactivity from about 0.2 Ci/ml to as low as about 2 mCi/ml. Residual radioactivity was almost exclusively due to Ru 106 and total alpha activity was only a few pCi/ml. Alternative methods of consolidating the solidified waste were evaluated using non-radioactive simulants. Best results were obtained by pressure sintering, which yielded essentially fully dense ceramics, e.g. titanate/titania ceramics with bulk densities as high as 4.7 g/cm cm, waste oxide contents as high as 1.2 g/cm cm, and leach resistance comparable to good borosilicate glass. Based on these results, a baseline process for solidifying high-level wastes was defined and approximate economic analyses indicated costs were not prohibitive. Additional tests have demonstrated that, if desired, operating conditions could be modified to allow recovery of Cs, and possibly other isotopes, during solidification of the remaining constituents of high-level waste. Preliminary tests have also shown that these materials offer promise for treating tank-stored neutralized wastes. (Auth) (LKH)

Density

Sr 90; Tc 99; Ru 106; Cs 137; Ce 144; Eu 154; Pu

SOLIDIFICATION; WASTES, HIGH-LEVEL; WASTES, LIQUID; ION EXCHANGE; SPENT FUELS; RESINS; RADIOACTIVITY; SINTERING; CERAMICS; TITANIUM; OXIDES; LEACHING; COSTS; WASTE TREATMENT; WASTE MANAGEMENT; LABORATORY STUDIES; METHODS

<637>
Hess, E.T., Jr., and J.H. Longo, Corporate Research Laboratories, Exxon Research and Engineering Company, Linden, NJ.

Confinement of Volatilized Ruthenium Oxides.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 487-490), 563 pp. (1979)

Fission product ruthenium poses unique problems in nuclear fuels reprocessing due to the volatility of ruthenium oxides. Results are presented from studies that demonstrate the ability of alkaline earth metal carbonates and calcium oxide to form thermodynamically stable ruthenium compounds and thus fix the volatile ruthenium oxides. Thermogravimetric experiments were done to establish the minimum reaction temperature of ruthenium oxides with alkaline earth carbonates such as $CaCO3$, $BaCO3$, and $SrCO3$. Results indicated that significant reaction occurs above 550 C. A series of trapping experiments to give information about the kinetics of the reaction of volatile ruthenium oxides with alkaline earth carbonates and calcium oxide established that the alkaline earth compounds such as $CaCO3$, CaO, and $BaCO3$ are effective for trapping volatilized ruthenium. An additional series of experiments done with volatilized ruthenium passed through granular $BaCO3$ confirmed the formation of alkaline earth metal ruthenates as the fixation process for ruthenium. The use of alkaline earth metal compounds as ruthenium sorbants presents other features advantageous for long term confinement, such as the ability to use them as coatings on ceramic substrates, or the stability of a compound like $BaRuO3$ under oxidizing conditions to 1000 C. (Auth) (DCW)

Two figures are presented which show thermogravimetric behavior of $CaCO3$ indicating the release of $CO2$ in the presence and absence of $RuO2$, and ruthenium concentration profiles in $BaCO3$ trapping beds.

SPENT FUELS; FUEL REPROCESSING; RUTHENIUM; CHEMISTRY; CHEMICAL PROPERTIES; FISSION PRODUCTS; LABORATORY STUDIES; CONTAINMENT; STABILIZATION, CHEMICAL; FIXATION

<638>
Mathers, W.G.

Some Properties of Hydrogen-Based Clinoptilolite.

TID-7648; Use of Inorganic Exchange Materials for Radioactive Waste Treatment, D.E. Janison, et al (Eds.), Proceedings of a Working Meeting, Washington, DC, August 13-14, 1962. U.S. Atomic Energy Commission Division of Technical Information, Washington, DC, (pp. 67-78), 238 pp. (TID-7648).(1963, January)

Sorption of Cs 137 and Sr 90 onto a column of clinoptilolite indicated some apparent anomalies. The columns sorbed more Cs and Sr than was predicted on the basis of 3 hour equilibrations and the 50% breakthrough capacities were the same for the two elements using de-ionized water. Also observed was a constant decontamination factor of 100 for rare earths. The volume of Cs 137 and Sr 90 containing radwaste was 8,200 liters. However, when Ottawa water was used the breakthrough capacities were reduced. Laboratory studies were initiated to determine if calcium, strontium, or complexing agents present in the river water were reducing the breakthrough capacities. The equilibrium capacity at 50% breakthrough as determined by sorption columns was 1.03 meq/g Sr and 1.10 meq/g Cs. The columns were eluted in an effort to understand the effect of acid treatment on loss of capacity. A three step elution cycle was run and after step one 91% and 0.5%, step two 19% and 0.06%, and step three 1.6% and 0.06% remained

<639> CONT.
on the columns. The test was run again with similar results. No clear cut interpretation of the data could explain the capacity loss. Concentration effects and calcium interference were also studied, as well as complexing agents. From the study it is certain that the calcium and strontium ions in the river water interfered, but the mechanism is not apparent. The lack of constancy of the mass-action coefficients and the selective interference exhibited by calcium suggest that chemically-different types of exchange site exist in hydrogen-based clinoptilolite. Mechanisms for exchange capacity loss after contact with nitric acid are not known. (NDW)

Waste Volume; Distribution Coefficient

Cs 137; Sr 90

SORPTION; BREAKTHROUGH DISTRIBUTION; CALCIUM; CESIUM; CLINOPTILOLITE; DECONTAMINATION FACTORS; ION EXCHANGE CAPACITY; LABORATORY STUDIES; FIELD STUDIES; ROCK-FLUID INTERACTIONS; WASTE DISPOSAL; WASTES, LIQUID; WASTES, INTERMEDIATE-LEVEL; STRONTIUM; SOLIDIFICATION; ENCAPSULATION

<639>
Matsuzuru, H., and A. Ito, Japan Atomic Energy Research Institute, Division of Environmental Safety Research, Takai-mura, Ibaraki-ken, Japan.

Effect of Dimension of Specimen on Amounts of Cesium 137, Strontium 90 and Cobalt 60 Leached from Matrix of Hardened Cement Grout.

Journal of Nuclear Science and Technology 15(4):296-301.(1978, April)

The relationship between the fraction leached and the ratio of surface area and volume (S/V) of a specimen has been studied in order to evaluate the safety of a cement composite for disposal. The leaching fraction of radionuclides was measured for the different S/V ratios. In a case of either Cs 137, Sr 90 or Co 60, the amount leached is directly proportional to the S/V ratio. It is in harmony with the empirical formula proposed here, which can be derived from the diffusion equation with a semi-infinite plane source model. The amounts of Cs 137, Sr 90 and Co 60 leached from a cement composite of 200-l drum size are estimated on the basis of the above relationship, using the data obtained with a cylindrical shape sample (8.5 cm diameter, 8.4 cm high). (Auth) (PAF)

CESIUM 137; STRONTIUM 90; COBALT 60; WASTES, LIQUID; SOLIDIFICATION; LEACHING; CEMENTS; GROUTING; WASTE VOLUME; DIFFUSION; EQUATIONS; DRUMS; MODELS, MATHEMATICAL

<640>
McCarthy, G.J., W.B. White, and D.E. Pfoertsch, Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Synthesis of Nuclear Waste Monazites, Ideal Actinide Hosts for Geologic Disposal.

Materials Research Bulletin 13:1239-1245.(1978)

Synthesis of monazite as a mineral waste form for geologic disposal of long-lived actinides has been successfully carried out on a laboratory scale. Monazite is potentially valuable for waste isolation because of its stability under a variety of geologic conditions. Simulated calcination, pelletization, and crystallization-sintering processes were carried out using seven combinations of nitrates of the lanthanides, Ca, Th, $(UO_2)^{++}$, $NH_4H_2PO_4$, and colloidal silica (DuPont Ludox-AS), fired at 1050 and 1200 degrees C for 4 and 48 hrs. X-ray diffractograms of all 8 firings of five of the compositions yielded only sharp, intense reflections of a monazite phase resembling $NdPO_4$. All observed reflections for these compositions could be indexed on refined monazite unit cells. Products of the other two compositons were multi-phase, with about 80 wt% being monazite. It was demonstrated that monazites with high concentrations of U or Th can be routinely synthesized from nuclear wastes. For high-Th monazites, Ca-RE or Si-P charge compensation can be used; for high-U monazites, only Ca-RE substitution appears to be effective. The availability of two solid solution mechanisms could be useful, as one type may be more stable under the geologic conditions of interest. (LKN)

LABORATORY STUDIES; URANIUM; THORIUM; WASTE DISPOSAL; STORAGE, GEOLOGIC; MONAZITE; WASTES, RADIOACTIVE; MINERALS; ACTINIDES; STABILIZATION, CHEMICAL; STABILIZATION, PHYSICAL; CALCINATION; SINTERING; NITRATES; METHODS

<641>
McDaniel, E.W., and H. W. Weeren, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Flowability of Bulk Solids Used in Shale Fracturing Grouts as Determined by the Jenike and Johanson Method.

ORNL/TM-6882; 18 pp.(1979, September)

The shale fracturing process for the permanent disposal of locally generated intermediate-level waste solutions is currently in use at Oak Ridge National Laboratory (ORNL). In this process, the waste solution is mixed with a solid blend of cement and other additives; the resulting grout is injected into an impermeable shale formation at a depth of 700 to 1000 ft. The existing facility was built in 1963 and has since been used to inject over 2.5 million gal of waste grout. Smooth operation of the facility requires that an even, uninterrupted flow of dry solids be delivered to a mixer at a rate of approximately 1200 lb/min. Some operational difficulties have been experienced with the system. Occasionally the solids flow from a bin was difficult to initiate or was irregular, particularly when the storage bin was nearly empty. A new shale fracturing facility is now being designed with a solids handling system that will eliminate the current problems. This report describes the tests made to obtain the necessary information about bulk-flow properties of the dry solids that are required for the facility modifications. The flow properties of fly ash, attapulgite, pottery clay, and a blended solid mix were determined in a Jenike and Johanson Flowfactor Tester. Tests with this unit involve the consolidation of a solid at a given pressure and the measurement of the force required to create a shear plane. The

<681> CONT.

procedure is repeated several times at different consolidating pressures to determine a locus of points that can be displayed graphically. The results indicate that a bottom-hopper opening of about 8 ft will be required on the new, bulk storage bins if mass flow is to be achieved. It was also found that the existing bins are quite unsuitable for either their present function (storage of dry mix) or their proposed function (storage of mix components). Since these bins have served moderately well for 18 years, the effect of aeration (not allowed for in these tests) is obviously major; these tests indicate only an upper limit to the size of the hopper opening that will be required. Pottery clay and attapulgite were found to have flow characteristics considerably inferior to blended solids. It is known, however, that attapulgite clay will flow from the existing bins, and modification of the bin bottoms to enlarge the openings does not appear to be justified. (Auth) (LKH)

SHALES; SEDIMENTARY ROCKS; METHODS; DESIGN; EQUIPMENT; WASTE DISPOSAL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WELLS, INJECTION; GROUTING; CEMENTS; FLY ASH; ATTAPULGITE; CLAYS; STEELS; PRESSURE EFFECTS; WASTE PROCESSING; LABORATORY STUDIES

<682>

McVay, T.N., R.L. Hamner, and H.P. Haydon, Oak Ridge National Laboratory, Oak Ridge, TN.

Fixation of Radioactive Wastes in Clay-Flux Mixes.

TID-7517 (Part 1B); Sanitary Engineering Aspects of the Atomic Energy Industry, Proceedings of a Seminar, Cincinnati, OH, December 6-9, 1955, (pp. 335-338).(1956, October)

Results are presented on the use of fired ceramic masses to fix radwaste. The objectives of primary importance in this program are: radioactive wastes are treated as received from the fuel reprocessing plant and if the activity is present they will be allowed to be self-sintered; materials added to the process solutions are to be cheap (clays, shale, silica, limestone, dolomite, fly ash, phosphate tailings and soda ash); temperature of firing should be low; the amount of clay or other diluents should be kept as low as possible to minimize the weight and volume of the final atomic garbage; and materials added should be capable of forming a slurry to minimize zone settling of the additives. The "Hope solution", 1.6 moles of aluminum nitrate and 0.2 mole of nitric acid, was used with different process solutions. The Hope solution was tagged with Sr 90, Ce 144, Pr 144, Cs 137, Ba 137, Ru 106, Rh 106, and mixed fission products. After such testing the two best solutions were limestone-sodium carbonate and shale. Leaching experiments were then conducted on these two. It was found that satisfactory firing occurred at 1050 F. These tests indicate there is a good possibility of fixing radioisotopes from waste solution in a ceramic body so the leaching effects will be minimal. (NDV)

Sr 90

CERAMICS; CONTAINMENT; FISSION PRODUCTS; FUEL REPROCESSING; LABORATORY STUDIES; LIMESTONES; SHALES; SILICA; WASTE TREATMENT; WASTES, RADIOACTIVE; SLURRIES

<683>

Mendel, J.E., Battelle-Pacific Northwest Laboratories, Richland, WA.

High-Level Waste Glass.

Nuclear Technology 32:72-87.(1977, January)

Glass is a good material in which to incorporate high-level radioactive waste (HLW) for permanent storage. HLW, a complex mixture of fission products and actinides, results from the reprocessing of spent power reactor fuel elements to reclaim uranium and plutonium. Processes for making low-temperature waste glasses (1050 C processing temperature) have been developed to the stage that they can be utilized in commercial reprocessing plants in the early 1980's. A representative low-melting waste glass formulation has been shown, in accelerated tests, to possess satisfactory thermal and radiation stability for many centuries of storage, and indications are that this stability will be maintained for longer times. The waste glass can be melted and stored in Type 304L stainless-steel canisters, although investigations of metals that may have increased high-temperature strength is continuing. A ceramic melting process that will permit manufacture of higher melting HLW glass, if this proves desirable, is also being developed. (Auth)

WASTES, HIGH-LEVEL; GLASS; VITRIFICATION; CONTAINMENT; WASTE STORAGE; ACTINIDES; FISSION PRODUCTS; THERMAL PROPERTIES; MELTING; CANISTERS; DESIGN; STEELS; LEACHING; RADIATION EFFECTS; DENSITY; ALPHA PARTICLES

<684>

Mendel, J.E., J.L. McElroy, and A.M. Platt, Battelle-Pacific Northwest Laboratories, Richland, WA.

High-Level Radioactive Waste Management Research and Development Program at Battelle Pacific.

CONF-760386; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, H.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 7, (pp. 93-107), 166 pp.(1976)

Solidified waste forms for the immobilization of high-level radioactive wastes from the commercial reprocessing of power reactor fuels, and processes for the reliable production of the waste forms are being developed at Battelle-Northwest. Current development is emphasizing silicate glass or glass-ceramic forms, produced in a two-step, calcination-melting process. Because glass will incorporate a wide variety of elements into its structure as network former or modifiers, it is believed it will prove very useful in the immobilization of radioactive waste elements. Depending on the waste composition, the glasses or glass-ceramics can contain over 25% (by weight) fission products and 2% actinides. (JHT)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL;

<(684)>

WASTE PRODUCTION

<(684) CONT.
ACTINIDES; CERAMICS; GLASS; CONTAINMENT; CALCINATION; FISSION PRODUCTS; WASTES, LIQUID; WASTE TREATMENT; FIXATION; SOLIDIFICATION; SILICATES; WASTES, SOLID

<(685)>
Merritt, W.F., Atomic Energy of Canada, Limited, Chalk River Nuclear Laboratories, Biology and Health Physics Division, Chalk River, Ontario, Canada.

The Leaching of Radioactivity from Highly Radioactive Glass Blocks Buried Below the Water Table: Fifteen Years of Results.

CONF-760310; STI/PUB/433; IAEA-SM-207/98; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 2, (pp. 27-35), 424 pp. (1976)

Test burials of high-level fission products incorporated into blocks of nepheline syenite glass and monitored for 15 years indicate that nuclear wastes may be safely stored in this manner below the water table in the waste management area of Chalk River Nuclear Laboratories, Chalk River, Ontario. The first set of blocks buried in August 1958 exhibited leaching rates too low to be readily measurable, so a second set containing greater proportions of uranium and corrosion products, with laboratory leaching rates ten times those of the first, were buried in May 1960. It was calculated that the amount of radioactivity leaving these blocks should be 150 times that of the initial test. Monitoring with tritium tracers has shown that ground water velocity past the blocks is 18 cm/d, which gives a volume of water based on the flow cross-section of the blocks of 39 l/d. Total Sr 90 leached over the past 15 years is 1.5 mCi, 90% of which was lost in the first two years. By 1976 the Sr 90 front was 33 m downstream from the disposal site. The leaching rate for the last 7 years has been roughly constant at about 1.6 x 10(E-8) uCi/l. The quality of glass used can be greatly improved, as evidenced by the first set of blocks, and the initially large loss of Sr 90 could be kept from the environment by pre-leaching. Lowering the surface/volume ratio by making larger blocks will also reduce leaching rates. A major factor in these test values was the low ground water temperature (mean about 7 degrees C.); at high water temperatures leaching increases considerably. (LKH)

Leaching Rate; Flow Rate; Hydraulic Velocity; Porosity

Sr 90

NEPHELINE; SYENITES; GLASS; WASTE STORAGE; WASTE MANAGEMENT; CONTAIN; WASTES, HIGH-LEVEL; WATER TABLE; SILICATES; STRONTIUM; LEACHING; GROUND WATER; IRRADIATION; FIELD STUDIES; FISSION PRODUCTS; MONITORING

<(686)>
Morgan, J.M., Jr. (Ed.), D.W. (Ed.) Jamison, and J.D. (Ed.) Stevenson, Johns Hopkins University, Department of Sanitary Engineering and Water Resources, Baltimore, MD.

Fixation of Radioactivity in Stable, Solid Media.

TID-7613; Report of the 2nd Working Meeting, Idaho Falls, ID, September 27-29, 1960. U.S. Atomic Energy Commission, Office of Technical Information, Washington, DC, Book 1, 782 pp. (1961, February)

Presented are the proceedings of a working meeting, which reviewed the progress made in transforming liquids and fixing radioactivity in a solid form. The major topics were: fluidized bed and radiant heat spray; pot calcination, rotary kiln, fixation in glass, and ion exchange; off-gas cleaning; materials of concentration; and characteristics of solids and final disposal. Abstracted separately were: fixation of low and intermediate activity in low melting inert media; calcination of zirconium fluoride waste; characteristics of some solids for fixation; waste fixation on minerals; ion exchange behavior of vermiculite-biotite; cesium-sodium sorption on clinoptilolite; mineral reactions pertinent to radwaste treatment and disposal; and radioactive waste fixation by coprecipitation and fusion. (NDV)

SORPTION; CALCINATION; CLINOPTILOLITE; COMPLEXES; DECONTAMINATION; EQUILIBRIUM CONSTANT; FISSION PRODUCTS; FLUIDIZATION; IMMOBILIZATION; ENCAPSULATION; ION EXCHANGE; MODELS; PRECIPITATION, CHEMICAL; WASTE TREATMENT; WASTES, RADIOACTIVE; WASTES, LOW-LEVEL; WASTES, HIGH-LEVEL; SOLIDIFICATION; WASTES, INTERMEDIATE-LEVEL; FIXATION

<(687)>
Navratil, J.D., L.L. Martella, C.H. Smith, G.H. Thompson, D.L. Cash, E.L. Childs, and L.J. Meile, Rockwell International, Rock Flats Plant, Chemistry Research and Development, Golden, CO.

Waste Management Analysis for the Nuclear Fuel Cycle-Parts 1 and 2: Progress Report for Period April 1 to September 30, 1977.

RFP-2749; 22 pp. (1978, October 23)

This progress report discusses the most recent work on actinide recovery and recycle preparation for waste streams and actinide recovery from combustible waste. The report is part of the effort to provide technology and design for fuel recycling facilities and to determine the economic and environmental impact of recycling. Completed during the time period was a preliminary evaluation of methods for the salt waste and waste water streams and recycle preparation problems. From a feasibility study done on removal of actinides from synthetic salt, bidentate organophosphorus extractant was most efficient. A combination of non-ionic and a strong base ion exchange resin gives the best results in removing detergents and anions from waste water. Evaluation of leaching and dissolution methods for recovering actinides from combustible waste (ash) was continued. The most promising methods to date are: reaction with cerium (IV) in nitric acid to solubilize carbon and actinide oxides; and fusion with carbonate-nitrate mixtures. Silica was a problem and pretreatment of the ash by refluxing with greater than or equal to 6M sodium hydroxide appears to eliminate the silica. More work will be done in these areas in the future. (NDV)

<647> CONT.
ACIDS; ACTINIDES; ASHES; COMBUSTION; DECONTAMINATION; COST BENEFIT ANALYSIS; LEACHING; DISSOLUTION; ION EXCHANGE; ACIDS, ORGANIC; WASTE TREATMENT; WASTES, LIQUID; WASTES, SOLID

<648>
Navratil, J.D., G.H. Thompson, and R.L. Kochen, Rockwell International, Rocky Flats Plant, Golden, CO.

Waste Management of Actinide Contaminated Soil.

CONF-781121; Science Underlying Radioactive Waste Management, Proceedings of a Symposium, Boston, MA, November 29-December 1, 1978, (pp. 22-23). 77 pp.(1978)

Waste management processes have been developed to reduce the volume of Rocky Flats soil contaminated with plutonium and americium and to prepare the contaminated fraction for terminal storage. The primary process consists of a wet-screening method which reduces the volume of contaminated soil by approximately 65%. The secondary process also employs wet-screening, but various reagents are added to the process water to aid in dispersing and decontaminating the soil. Also, attrition scrubbing and flotation are used in conjunction with wet-screening to decontaminate an additional 10-25% of the soil. The actinide-rich fraction is then vitrified for terminal storage. The results of testing the primary and secondary soil decontamination process on a laboratory scale will be presented. Vitrification methods for the contaminated soil fraction will be discussed to include high-temperature calcination with or without glass-forming reagents. Tabulation of properties, such as density and solubility, of the final calcined and glass soil fractions will also be given. (Auth)

Abstract is complete text. The paper was not published in the Proceedings.

VOLUME REDUCTION; PLUTONIUM; AMERICIUM; DECONTAMINATION; ACTINIDES; SOILS; VITRIFICATION; CALCINATION; METHODS

<649>
Nelson, J.L.

Hanford Mineral Exchange Program.

TID-7644; Use of Inorganic Exchange Materials for Radioactive Waste Treatment, D.K. Jamison, et al. (Eds.), Proceedings of a Working Meeting, Washington, DC, August 13-14, 1962. U.S. Atomic Energy Commission, Division of Technical Information, Washington, DC, (pp. 83-114), 238 pp. (TID-7644).(1963, January)

Some of the experience concerning mineral exchange at Hanford is discussed. Studies on alkaline supernatant solutions from the high-level Purex tanks included a hot cell run with 200 gal of actual waste. The results were in good agreement with laboratory data. Plans for waste management changed from using bulk calcination of waste to the separation of the long-lived heat emitters cesium, strontium, and rare earths from the wastes. The waste is then put into interim storage for 2 to 10 years to decay and solidified to a salt cake by in-tank solidification methods. A new process for handling acid high-level waste by using inorganic exchangers is outlined fully. The waste is called FTW, which stands for formaldehyde-treated-wastes. Cesium is extracted from the waste by clinoptilolite columns. Oxalic acid is used to remove some of the zirconium and niobium taken up by the mineral. Ammonium hydroxide is then used to rinse the column and any cesium eluate is run through a zeolite exchanger. The wastes are finally washed, dried and sealed for storage in water-filled bays. The optimum flow rate is about 2 to 4 column volumes/hour. Concentration of cesium in the columns has a small effect on the breakthrough curve. Lower temperatures at loading are desirable for limited breakthrough. After many experiments it is probably worthwhile to dilute the wastes. For larger size particles of exchange material the kinetics are poorer. Decontamination factors for cesium, strontium, and ruthenium are greater than 9 x 10(E+4), 2 x 10(E+3), and 1.4 x 10(E+5), respectively. A number of elution studies were done on the system using different eluants. Studies using other exchangers were also made. (NDW)

Flow Rate

SORPTION; BREAKTHROUGH DISTRIBUTION; CLINOPTILOLITE; DECONTAMINATION; DECONTAMINATION FACTORS; FIELD STUDIES; ION EXCHANGE; LABORATORY STUDIES; PLANTS, WASTE TREATMENT; CESIUM; HOT CELLS; PUREX PROCESS; CALCINATION; SOLIDIFICATION; ACIDS; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, LIQUID; STRONTIUM; ZIRCONIUM; ZEOLITES; ENCAPSULATION

<650>
Nieschmidt, E.B., and S.H. Vegors, Jr., EG and G Idaho, Incorporated, Idaho Falls, ID.

Selection of a Criticality Monitoring Technique for a Transuranic Waste Incinerator.

TREE-1277; 38 pp.(1978, June)

Active and passive monitoring techniques were studied and compared for determining the amount of fissile material in transuranic waste before and after the waste is incinerated. The results of this study indicate that the photon interrogation method is the best technique for criticality monitoring of transuranic waste flowing into and out of the slagging pyrolysis incinerator. Other counting techniques investigated included gamma ray, neutron, x-ray, alpha, thermal neutron, fission neutron, and fast neutron. Passive methods were ruled out because: 1) the low energy and/or limited penetrating ability of the radiation emitted by the wastes during decay makes the amount of radiation observed a poor indicator of the amount of fissile material present; 2) the random distribution of shielding materials (iron, lead, etc.) in the waste will severely attenuate the amount of radiation reaching the detectors and make mathematical corrections impractical; 3) some of the waste emits radiation due to activity not associated with the fissile material, compounding the difficulty of measuring emissions from the fissile material. Investigation of active methods shows that: 1) fast neutrons are better than slow neutrons because they are less affected by

<650> CONT.
the matrix in which the TRU waste is embedded and because they induce fissions in both fertile and fissile components of the TRU, providing an automatic safety margin in measurements; 2) photon interrogation is preferrable over neutron because the former allows use of prompt fission neutrons (as opposed to delayed fission neutrons) as a signature for fission, increasing sensitivity a thousandfold; 3) effects of the matrix material surrounding the TRU are less when using gamma rays than when using neutrons as interrogating particles. Recommendations include performance of a cost analysis and installation schedule for a linear accelerator at the Radioactive Waste Management Complex, development of calibration procedures for the types of TRU and radwaste present at INEL, and experiments to select the most desirable detector mechanism and configuration. A schedule of further efforts for FY 1978 is given. (LKH)

THEORETICAL STUDIES; MONITORING; WASTES, TRANSURANIC; WASTES, SOLID; INCINERATION; FISSION; NEUTRONS; RADIATION, GAMMA; RADIATION DETECTORS; RADIOACTIVITY; RADIOACTIVE DECAY; WASTE MANAGEMENT; METHODS

<651>
Fanter, H.B., and H.E. Mason, Nuclear Assurance Corp., Atlanta, GA.

Experience in Volume Reduction, Packaging, Transportation, and Disposal of Non-Fuel Bearing Hardware Resulting from Light Water Reactor Operations.

Management of Low-Level Radioactive Waste, M.W. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 6, (pp. 551-562), 1218 pp. (1979)

In order to conform to various shipping and burial regulations specifying allowable dose rates and isotopic content, activities of the non-fuel bearing components resulting from light-water reactor operations were calculated by the computer code ORIGEN, based on reactor operating parameters, cool times, and ASTM specifications. These data were used to evaluate various packaging configurations and/or shipping casks for conformity to the regulations and disposal site license constraints. Because of the high transportation and burial charges, the overall cost of disposing of this type of waste is highly dependent on the number of shipments. The Nuclear Assurance Corp. has developed equipment to cut bulky items and reduce the packing volume of the individual components tenfold. Equipment design and development also considered remote, underwater environment, physical characteristics of the irradiated material, i.e., zirconium embrittlement, the need to minimize pool floor space and interference with other pool activities, operating convenience, and maintenance requirements. Experience has shown that nonfuel hardware can be cut and packaged underwater in a safe and practical manner. Tools must be carefully designed and tested in order to keep reliability to a maximum, since the tools must be removed from the pool and placed in a controlled area for maintenance and repair. Extra time spent in the testing stages will be more than made up for by time saved during actual operations. Personnel problems associated with non-fuel bearing components disposal have been minimal. Usually the personnel dose is 150 mrem or less, based on 12 hours/day and a 6 days/week work schedule. However, operation fatigue can become a problem after several weeks of constant operations. (Auth)

COMPUTER CODES; COMPUTER PROGRAMS; EVALUATION; PACKAGING; CASKS; COSTS; VOLUME REDUCTION; EQUIPMENT; DESIGN; RADIATION DOSE; WASTES, RADIOACTIVE; WASTES, SOLID; EXPOSURE, OCCUPATIONAL

<652>
Pechin, W.H., D.E. Blanco, R.C. Dahlman, R.C. Finney, R.S. Lindauer, and J.P. Witherspoon, Oak Ridge National Laboratory, Metals and Ceramics Division, Oak Ridge, TN; Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "As Low As Practicable" Guides - Fabrication of Light-Water Reactor Fuel from Enriched Uranium Dioxide.

ORNL/TM-4902; 208 pp. (1975, May)

A study was made to determine the cost and effectiveness of radwaste and chemwaste treatment systems for 2 model enriched-uranium, light-water reactor fuel fabrication plants, and to determine the dose commitment of released radionuclides to the environment. One model plant uses the wet ammonium diuranate (ADU) process, the other the dry direct conversion (DC) process. The plants are assumed to each process 1500 metric tons of U per year (5 tons/day for 300 days). Th 231, Th 234, and Pa 234 daughters are in equilibrium with the UF6 feed (4% U 235). Four conceptual cases were modeled: a base case representing the lowest cost and technology used in some current plants, and successive cases in which treatment equipment is added. In advanced case studies, treated water, recovered nitrogen oxides (as nitric acid), and ammonia are recycled in the plants; in all cases, solid radwastes and chemwastes are stored on-site or shipped to commercial burial grounds. Model plants are evaluated for a midwestern site and a southeastern coastal site, using typical meteorological patterns and population densities for each. The annual amounts of radioactive materials released (the source term), the capital, annual and contribution to power costs, and the radiological impact are calculated for each case. The annual radwaste treatment costs are compared with the dose commitment from released radionuclides; comparison with commitments from liquid effluents are on two bases: 1) annual individual total body dose from aquatic pathways after dilution in a 15 cfs stream, and 2) after the 15 cfs stream flows into a 1300 cfs stream (at the midwestern site) or into an estuary (at the coastal site). Estimated maximum total body doses do not exceed 1 mrem/yr within 0.5-1 mi of the Case 1 plant at either site. Average individual dose is lower at the midwestern site due to meteorological factors, but the population dose is higher due to the denser population. Capital costs of the treatment systems modeled range from $0.586 million for DC Case 1 to $8.99 million for ADU Case 4, or up to about 18% of the $36 million which is

<652> CONT.
taken as the cost for the base plant. Annual costs are $202,000-$2,515,000/hr, corresponding to contribution to power costs of 5.5 x 10(E-4) and 6.8 x 10(E-3) mills kWhr, respectively. These values are less than 0.1% of an estimated total power generation cost of 7-10 mills/kWhr. (LKM)

Methodology used in estimating costs and radiological doses, detailed calculations, and tabulations are presented in Appendix A and CNSL-8992.

COST BENEFIT ANALYSIS; WASTE TREATMENT; FUEL FABRICATION; DOSE COMMITMENTS; MODELS; MODELS, MATHEMATICAL; RECYCLING; WASTES, LIQUID; EFFLUENTS, LIQUID; STREAMS; METEOROLOGY; EFFLUENTS; WASTES, RADIOACTIVE; WASTES, NONRADIOACTIVE; THEORETICAL STUDIES

<653>
Paltonen, E.K., J.W. Heinonen, and J. Kauni, Technical Research Center of Finland, Reactor Laboratory, Espoo, Finland; Oy Finnatom Ab, Helsinki, Finland.

Incorporation of Radioactive Wastes from Nuclear Power Plants into Concrete and Bitumen.

CONF-760310; STI/PUB/433; IAEA-SM-207/5; Management of Radioactive Wastes from the Nuclear Fuel Cycle, Proceedings of an IAEA Symposium, Vienna, Austria, March 22-26, 1976. International Atomic Energy Agency, Vienna, Vol. 2, (pp. 97-112), 428 pp.(1976)

A study was conducted to assess the suitability of cementation and bitumenization processes for the solidification of liquid wastes from nuclear power plant operations (excluding fuel reprocessing wastes) and, in particular, their applicability to wastes from the first Finnish reactor, a VVER-440, in Loviisa. The conclusion of the investigators was that the technology necessary for the solidification of wastes is still under development. There is a need for standardization of terms, concepts, units, and tests, and for the specific identification of those properties of products which are significant for storage or final disposal. Of the two processes considered, no preferability of one over the other was determined. Both are applicable under certain conditions. Bitumenization products may be better suited for temporary on-plant storage due to their lower volumes. Characteristics of the two types of products and their significance from the standpoint of process, transportation, storage, and disposal are listed in tabular form. (LKM)

CEMENTS; CONCRETES; BITUMINIZATION; WASTE MANAGEMENT; COST BENEFIT ANALYSIS; NUCLEAR FACILITIES; EVALUATION; VOLUME REDUCTION; THEORETICAL STUDIES; WASTES, RADIOACTIVE; WASTES, LIQUID; SOLIDIFICATION; WASTE TREATMENT

<654>
Pierce, R.E., Portland General Electric Company, Portland, OR.

Retrofitting Trojan Nuclear Plant's Spent Resin Transfer System.

Management of Low-Level Radioactive Waste, N.W. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 6, (pp. 581-592), 1214 pp.(1979)

The spent resin slurry transport (SRT) system at the Trojan Nuclear Plant operated by Portland General Electric Co. is one of the most advanced systems of its type in the nuclear industry today. The new system was developed following repeated clogging of the original system's single water-driven spent resin eductor during initial plant startup testing, and realization that larger quantities of contaminated water would be generated if the eductor-induced slurry transport scheme were to be pursued further. In the retrofitted SRT system, the original resin transfer eductor and SRT pump were removed and replaced by a single recessed-impeller centrifugal pump to alleviate persistent clogging problems occurring in the spent resin storage tank (SRST) suction line and SRT pump strainer. The new pump supplies the power to transfer the resin slurry directly from the SRST rather than depending upon induced suction power generated from within the resin transfer eductor. This obviated the need for large quantities of strained water required to develop the active and suction power of the eductor. The removal of the resin transfer eductor also precludes the need for extensive flow instrumentation required to optimize the resin transfer eductor's performance. The installation of the new pump does not affect the closed-loop process recirculation line, which transfers the resin slurry past the spent resin waste packaging room and back to the SRST. Use of the new SRT pump also precludes the generation of contaminated backflush water because the large SRST suction strainer formerly used is not required. The resin agitation generated by the fixed sparagers mounted within the SRST was not changed during the retrofit program because only the flow media were changed from strained water to resin slurry. Added features of the new system are a new resin level indicator, which, when combined with a new dewatering strainer, will afford the plant operators greater flexibility in obtaining optimized water-to-resin ratios and a means by which desirable slurry ratios may be duplicated. Streamlining of the SRT system's piping layout and repositioning of manual remote valve operators results in lowered overall personnel exposure since the system now drains by gravity and crud traps have been minimized. (Auth) (LKM)

RESINS; EQUIPMENT; SLURRIES; WASTE PROCESSING; WASTES, ORGANIC; WASTES, RADIOACTIVE; WASTE MANAGEMENT; INSTRUMENTATION; PUMPS; TANKS; WASTE TREATMENT

<655>
Plodinec, M.J., Savannah River Laboratory, Aiken, SC.

Evaluation of Cesium 137 Sorbents for Fixation in Concrete.

DP-1440; 25 pp.(1977, February)

The purpose of this study was to determine if the Cs leachabilities of NAC and I-P concrete/waste mixes could be significantly reduced. Alternative Cs sorbents and processing methods were studied, including

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WASTE PRODUCTION

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Linde AW-300, AW-500, 13-X, and SK-40, Norton Zeolon 200, 500, and 900, clinoptilolite, and vermiculite. The important parameters studied for each sorbent were Cs sorption kinetics, compressive strength of concretes containing the sorbent, and leachability. Initial tests indicated that SK-40 and 13-X adsorbed Cs very poorly, and SK-40 was difficult to separate from the aqueous phase. Tests using a concentrated eluate solution, 0.019M Cs2CO3 in 1.75M Na2CO3, passed through a 100 ml column of each sorbent at 1 column volume(CV)/hr were performed. AW-500 and Z-900 were the best sorbents; each showed 100% retention of Cs 137 after 70 CV of eluate and 60% retention after 80 CV of eluate from Duolite ARC-359 ion exchange resin columns. Z-500 had 100% retention of Cs 137 for only 1 CV, and none at all after 8 CV. Vermiculite showed no retention after 2 CV. Compressive strength tests using mixtures of simulated sludge and sorbent incorporated into HAC and I-P concretes showed that in the absence of sludge, AW-500 and Z-900 have relatively little effect on the strength of the concrete (compressive strength 5400 psig versus 6900 psig without sorbent). Z-200, AW-300, and vermiculite severely weakened the concrete (compressive strengths 1000-1400 psig). In the presence of sludge, AW-500 and Z-900 actually strengthened the concrete, vermiculite weakened it badly, and the other sorbents had little effect. Compressive strengths of HAC and I-P concretes with 30 wt% simulated sludge and no sorbents were 1200 and 2300 psig, respectively. Addition of 10 wt% Z-900 raised these values to 2100 and 3200 psig; 10 wt% AW-500 yielded 2000 and 3200 psig, respectively, while vermiculite reduced strengths to 800 and 110 psig, respectively. Values for the other sorbents were 1700-1900 psig for the HAC mix and 2300-3100 psig for the I-P. Based on a lower limit of 1000 psig for safe handling, concrete waste forms containing up to 40 wt% simulated sludge can be made. Cesium leaching tests on HAC and I-P concretes containing 2 wt% of each sorbent gave values of 5.7-17.9% Cs leached after 28 days in HAC, and 3.8-9.3% leached from I-P. HAC concretes containing AW-500 and Z-900 lost 5.7% and 6.0% Cs, respectively; losses from I-P concretes were 4.6% and 3.9%, respectively. Vermiculite that had been heated to collapse its lattice around Cs 137 gave the most leach resistant concrete. However, it sorbed Cs slowly, and the resulting concrete was very weak. Addition of silica gel to concrete to react with free calcium had no effect on Cs leachability. In all tests, Linde AW-500 and Norton Zeolon 900 were found to be the most acceptable Cs 137 sorbents. Sorption kinetics were acceptable, both strengthened the concrete, and both gave relatively leach-resistant concrete. (Auth) (LKH)

Leaching Rate; Compressive Strength

Cs 137

LEACHING; CONCRETES; CESIUM 137; SORPTION; KINETICS; STRENGTH, COMPRESSIVE; CLINOPTILOLITE; VERMICULITE; ADSORPTION; SLUDGES; SILICA; CALCIUM; LABORATORY STUDIES; WASTES, RADIOACTIVE; IMMOBILIZATION

<656>
Plodinec, M.J., Savannah River Laboratory, Aiken, SC.

Development of Glass Compositions for Immobilization of SRP Waste.

DP-1517; 34 pp.(1979, February)

This study was undertaken to improve the processing characteristics of SRP waste glasses, without altering glass quality. Previous studies of sludge composition effects on glass properties found that 1) Al increases the viscosity and tendency to rapidly devitrify, 2) Fe reduces viscosity, but can form insoluble phases; and 3) other sludge components have little effect on viscosity, but Ni and U retard vitrification. Initial glass development was directed toward demonstrating the feasibility of waste immobilization in glass, leading to the development of a glassformer composition called Frit 18. Frit 18 could not successfully immobilize more than 25 wt% of high-aluminum sludges, so a new composition, Frit 21, was developed which could immobilize up to 35 wt% of these sludges, and adopted as the reference glass composition. Based on data obtained during this study, an improved frit, designated Frit 411, was selected for more extensive tests. Advantages of Frit 411 over Frit 21 include: Frit 411 sludge melts are much less viscous because of a higher Li/Na ratio; Frit 411 can dissolve more ferrite-spinel than Frit 21 because the TiO2 content is much lower; and Frit 411 can dissolve more U3O8 or PuO2 than Frit 21 (up to 7 wt% PuO2, or about 8X the amount expected in SRP waste glasses). Viscosities of Frit 21 and Frit 411 glasses containing 25 wt% simulated sludge were measured at 950 C, 1050 C, and 1150 C; viscosities were as follows: Frit 21/composite sludge - 273.0, 87.4, and 38.1; Frit 411/composite sludge - 98.6, 33.6, and 15.1; Frit 21/high-Al sludge - 1240.0, 307.0, and 107.0; Frit 411/high-Al sludge - 340.0, 117.0, and 50.9; Frit 21/high-Fe sludge - 108.0, 28.1, and 11.1; Frit 411/high Fe sludge - 29.6, 12.6, and 7.0; Frit 21/average sludge - 217.0, 58.0, and 21.9; Frit 411/average sludge - 44.2, 18.5, and 9.5. Average sludge is the arithmetic average of compositions sampled prior to this work. Composite sludge composition is a weighted average reflecting the relative amounts of sludge in sampled SRP tanks. For a given sludge the Frit 411 melt will reach the same viscosity about 100 C lower than the Frit 21 melt. Resistivities for Frit 411 melts are lower than for Frit 21 melts. Ruthenium solubilities in both frits were 0.02 wt%. Further testing is needed to determine the long-term thermal stability and leach rate of these two glasses; Frit 411 will be tested in large-scale experiments at Battelle Pacific Northwest Laboratories. (Auth) (LKH)

See also DP-781121, (pp. 31-35).

Viscosity; Resistivity

GLASS; VITRIFICATION; SLUDGES; WASTES, RADIOACTIVE; VISCOSITY; IMMOBILIZATION; RESISTIVITY; CHEMICAL COMPOSITION; THERMAL PROPERTIES; LABORATORY STUDIES; IRON; ALUMINUM; TITANIUM; SPINEL; NICKEL; URANIUM; PLUTONIUM

<657>
Plodinec, M.J., Savannah River Laboratory, Aiken, SC.

Viscosity of Glasses Containing Simulated

<657> CONT.
Savannah River Plant Waste.

DP-1507; 23 pp. (1978, August)

This report examines the effects of various sludges on the viscosity of glasses prepared from two frit candidates; simulated sludge loadings and compositions spanned the expected range for SRP waste glass. Glass melts were from two borosilicate glass frits, Frits 18 and 21. Frit 21 differs from Frit 18 by the substitution of 4 wt% Li2O for 4 wt% Na2O. Glasses made from Frit 21 contained either 25-45 wt% simulated high-iron sludge, 10-45 wt% simulated high-aluminum sludge, 25-45 wt% average sludge, or 25-35 wt% composite sludge; contents of sludge types for Frit 18 glasses tested were 25-45 wt%, 25-45 wt%, 25-45 wt%, and 25 wt%, respectively. Average sludge is the arithmetic average of compositions sampled prior to this work. Composite sludge is a weighted average reflecting the relative amounts of sludge in sampled SRP tanks. In amounts up to 30 wt%, high-Fe sludge decreased the viscosity of the glasses made with either frit relative to pure frit; larger proportions of this sludge increased the viscosity of the melt and caused non-Newtonian behavior. High-Al sludge greatly increased the melt viscosity; as little as 15 wt% in Frit 21 caused the viscosity to exceed 50 poise for all temperatures below 1200 C. Frit 18 is not suitable for processing this sludge below 1150 C. Viscosities of melts containing up to 30 wt% average sludge were approximately the same as those of pure frit. Above 30 wt%, glasses containing this sludge were non-Newtonian. Melts containing composite sludge were more viscous than pure Frit 21; as the amount of sludge increased, the viscosity increased at any given temperature relative to pure frit. At 35 wt%, composite sludge was non-Newtonian. Changing any frit component by 1 wt% had no significant effect on viscosity; therefore, a 1 wt% variability in any frit component can be tolerated. The results show that Frit 21 is clearly better than Frit 18 in terms of its low viscosity. However, further reductions in viscosity are desirable, especially for glasses containing high-Al sludge. (Auth) (LNH)

Viscosity

GLASS; VITRIFICATION; VISCOSITY; SLUDGES; WASTES, RADIOACTIVE; LABORATORY STUDIES; IRON; ALUMINUM; THERMAL PROPERTIES; IMMOBILIZATION; LITHIUM; SODIUM

<659>
Plodinec, M.J., DuPont Company, Savannah River Laboratory, Aiken, SC.

Development of Glass Compositions for Immobilization of Savannah River Plant Waste.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 31-35), 563 pp. (1979)

Two processes for solidification of high-level radioactive wastes from the Savannah River Plant (SRP) are described. Method 1 is in-can melting where the glass frit and calcined waste are vitrified in a canister. The melting temperature must produce fluid with a viscosity of less than or equal to 200 poise. Method 2, joule-heated melting, is a process where glass frit and calcine are fed continuously to a ceramic-melter, vitrified, and poured into the primary storage container. The viscosity of the fluid must be less than or equal to 50 poise at the melting temperature for an acceptable product and must have a viscosity of less than or equal to 500 poise for smooth pouring. In general SRP sludges are high in either aluminum or iron and this condition causes processing problems. Sludges containing large amounts of Fe2O3 produce crystals of a ferrite-spinel type when vitrified. At high concentrations these crystals cause the melts to become non-Newtonian pseudoplastic fluids; at low concentrations the crystals form a layer at the bottom of the joule-heated melter disrupting its smooth operation. Sludges containing large amounts of Al2O3 create extremely viscous melts, which readily devitrify to alkali aluminosilicates. To handle the extremes in composition four methods are suggested: increasing the melt temperature; reducing the amount of sludge; blending sludges; and changing frit composition. The most important glass composition variables on these processes are: TiO2 content, molar alkali-to-silica ratio, and molar ratio of Li2O to total alkali. (NDV)

See also DP-1517.

CALCINE; GLASS; IMMOBILIZATION; IRON COMPOUNDS; ALUMINUM COMPOUNDS; METHODS; SLUDGES; TEMPERATURE; TITANIUM COMPOUNDS; VITRIFICATION; SOLIDIFICATION; WASTES, HIGH-LEVEL; WASTES, LIQUID; VISCOSITY; LITHIUM COMPOUNDS; LABORATORY STUDIES

<659>
Rankin, W.W., and J.A. Kelley, Savannah River Laboratory, Aiken, SC.

Microstructures and Leachability of Vitrified Radioactive Wastes.

Nuclear Technology 41:373-380, (1978, December)

Metal oxide precipitates (primarily iron oxide compounds) will form in the glass matrix of some compositions of vitrified nuclear waste, during cooling of the melt, whenever solubility limits are exceeded. These precipitates, containing part of the cesium and strontium radionuclides from the waste, are more resistant to leaching by water than the as-cast glass matrix. Some of the glass matrix compositions devitrify during heating for 1 month at 600 C with the formation of equal amounts of NaAlSiO4 (nepheline or carnegieite) and (Ca, Na) (Mg, Fe, Mn) Si2O6, plus a small amount of CaFe14O25. The leachability of devitrified glass can be up to 100 times greater than the leachability of as-cast glass. The appearance and structure of the metal oxide precipitates are unaffected by the temperature conditions that caused devitrification of the glass matrix. The metal oxide precipitate particles are less leachable in water than any of the phases in the devitrified matrix. (Auth)

Leaching Rate

Sr 90; Cs 137

<659>

WASTE PRODUCTION

<659> CONT.
WASTES, HIGH-LEVEL; GLASS; VITRIFICATION; METALS; OXIDES; IRON COMPOUNDS; SOLUBILITY; PRECIPITATION, CHEMICAL; PHYSICAL PROPERTIES; CESIUM; STRONTIUM; LEACHING; WATER; THERMAL PROPERTIES; NEPHELINE; SILICATES; DEVITRIFICATION; ALPHA PARTICLES; LABORATORY STUDIES

<660>
Bindfleische, J.A., Allied Chemical Corporation, Idaho National Engineering Laboratory, Idaho Falls, ID.

Stabilization of Simulated Commercial Waste Calcine.

ICP-1087; 25 pp. (1976, June)

A fluidized-bed calcination process is being developed at the Idaho Chemical Processing Plant (ICPP) for conversion of high-level liquid radioactive wastes to granular solids. Pilot-plant studies with simulated waste have yielded a volume reduction factor of 20. Candidate waste forms for storage at a retrievable surface storage facility or federal repository include glass or glass-ceramics, sintered glass-ceramics, calcine incorporated in metal matrices, or stabilized calcine. Calcine canisterization or incorporation into a metal matrix requires stabilization (removal of water and nitrates) before canister storage. The fluidized-bed process is ideal for this stabilization. Experiments were conducted with a 2-inch diameter quartz laboratory calciner using calcine generated from a mixture of high- and intermediate-level wastes. The rate of denitration and the residual nitrate and water content were determined as a function of fluidizing velocity, bed temperature, and initial water and nitrate content of the calcine. Nineteen runs were made, 17 using simulated Allied-General Nuclear Services calcine and 2 using simulated ICPP zirconia calcine. The main variables affecting stabilization are temperature and initial nitrate content of the solids. Batch fluidized-bed denitration at temperatures of 600-700 C and fluidizing velocities of 0.2-0.5 ft/sec produced stable calcine with residual nitrate and water contents less than 0.02 wt% and 0.006 wt%, respectively. The highest initial nitrate concentration was 0.3 wt% for zirconia calcine and 9-10 wt% for simulated commercial calcine. Ruthenium release was insignificant; no appreciable Cs release occurred below 700 C, though significant amounts were evolved at 750 C. Maximum estimated canister pressures if all residual nitrate and water decomposed to NO, N2, and O2 are 62 psi for commercial calcine at a mean temperature of 525 C, 25 psi for ICPP calcine at 150 C, and 66 psi for commercial calcine in a metal matrix at 300 C. (Author) (LKM)

Cs; U; Ru; Sr

CALCINE; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; STABILIZATION, CHEMICAL; CONTAINERS; WATER; DENITRATION; WASTES, LIQUID; RUTHENIUM; CESIUM; CALCINATION; VOLUME REDUCTION; NITRATES; TEMPERATURE

<661>
Boss, W.A., D.J. Bradley, L.B. Bunnell, W.J. Gray, T.H. Katayama, G.H. Bellinger, J.E. Mendel, F.P. Roberts, R.P. Turcotte, J.W. Wald, W.E. Weber, and J.H. Westsik, Jr., Battelle-Pacific Northwest Laboratories, Richland, WA.

Annual Report on the Characterization of High-Level Waste Glasses.

PNL-2625; 55 pp. (1978, June)

A major goal of the Waste Fixation Program at the Pacific Northwest Laboratory is to characterize waste forms and processes for solidification of high-level radioactive wastes. A set of experiments was conducted on the effects of major (plus or minus a factor of 2) variations in the concentrations of principal oxides in a glassy waste. It was found that these major variations do not grossly affect the leach rates of the glass. Impact and strength tests were conducted on samples of nonradioactive glass produced in both laboratory- and engineering-scale equipment and were compared to commercial soda-lime and fused silica glasses. The impact tests showed that the waste glasses compared well and in general produced slightly less fine particulate than either commercial glass. Strength tests showed that waste glasses had 60% of the strength of the soda lime glass. Major differences (100X) were obtained in leach testing experiments by varying testing methods. Two leach test parameters which appear to be important are sample form (pellet vs. powder) and acid type. Leach tests of highly devitrified samples showed that leach rates of Cs, Sr, and U are increased up to 10 times and that Zn leach rates are reduced by nearly 200 times with devitrification. High temperature (250 and 350 degrees C) leach tests showed that glass is comparable to other ceramic materials and that all materials corrode at accelerated rates. In water, the glass structure is rapidly rearranged to a crystalline structure and Cs and Pb tend to remain bound in the solid. A comparison of the behavior of 8 reference glasses was made utilizing actual high-level waste. The results of these tests supported the extensive use of simulated waste glasses and demonstrated that glass can tolerate unanticipated differences in composition. All characterization results indicated that glass is a good waste form. (JMT)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; GLASS; LEACHING; LABORATORY STUDIES; WASTE PROCESSING; WASTE STORAGE; RADIATION EFFECTS; SOLIDIFICATION

<662>
Roy, D.M., R. Dainon, B.E. Scheetz, D. Wolfe-Confer, and F. Asaja, The Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Role of Admixtures in Preparing Dense Cements for Radioactive Waste Isolation.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 861-866), 563 pp. (1979)

Water, in the preparation of cement mixtures, serves to maintain the fluidity of the

<662> CONT.

original mixture and to enable the hydration reaction to progress causing setting and hardening. However, too much water will occupy empty pores and reduce strength and other mechanical properties of the concrete. Admixtures reduce the undesired water. In this study various polyelectrolytes containing carbocyclic, heterocyclic aromatic, or aromatic amine type compounds were investigated as possible additives. The effectiveness of such admixtures is related to their properties in solution and at the solution-cement solid interface. The properties measured included surface adsorption, surface potential, slurry properties. Kinetics of the hardening process as well as properties of the hardened composites were also studied. (DPV)

CEMENTS; PHYSICAL PROPERTIES; ENCAPSULATION; SOLIDIFICATION; WASTE TREATMENT; LABORATORY STUDIES

<663>

Foy, R., The Pennsylvania State University, Materials Research Laboratory, University Park, PA.

Science Underlying Radioactive Waste Management: States and Needs.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 1-20), 563 pp.(1979)

The Keynote Address for this Symposium describes scientific data and arguments which are needed to form a proper base for the entire technological system for nuclear waste management. Attention is focused on the immobilization at the source, i.e., the "waste form" portion of the system. The technical system constitutes a major, but not necessarily determinative, component of the national plan (system) for radioactive waste management (RWM). This technical RWM system can be divided into four subsystems: Storage; Radionuclide Immobilization; Isolation; Isolation and Host Emplacement. It is essential to consider the interactions among these four subsystems in designing any system. Of these, the first and last have not been given much consideration; their role is outlined in this paper. Of the middle subsystems, up to 1978 the system designers have relied essentially exclusively on the Isolation subsystem (i.e., removal from the biosphere in a mine or seabed, preventing migration of ions, etc.). History has shown that technically this strategy has proved unacceptable. In the last two years the role of the solid waste form and thence the entire Radionuclide Immobilization subsystem has assumed a much more significant role. Recent research results give grounds for hope that this subsystem may carry most of the burden of proof of acceptable and verifiable removal of radionuclides from the biosphere. While glass forms dominate most engineering programs, research on alternative solid forms is taking place. Alternative waste forms under investigation include: 1. Use of "stable" minerals as models of most resistant rocks; 2. Polycrystalline ceramic assemblages; 3. Cement-composites and grouting technology; 4. Ceramic, glass, and gel sponges; 5. Hot isostatic compaction; and 6. Chemical vapor deposition (CVD) coated ceramics. A list of topics which requires further research is given. (Auth) (WAF)

REVIEWS; WASTE MANAGEMENT; WASTE FORM; RADIONUCLIDES; IMMOBILIZATION; GLASS; MINERALS; METALS; CERAMICS; CEMENTS; GROUTING; COMPACTION; WASTES, SOLID; LEACHING; WASTES, RADIOACTIVE; WASTE TREATMENT

<664>

Rudolph, G., and H. Kester, Kernforschungszentrum Karlsruhe GmbH, Karlsruhe German Federal Republic.

Immobilization of Strontium and Cesium in Intermediate-Level Liquid Wastes by Solidification in Cements.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 467-470), 563 pp.(1979)

This paper describes strontium and cesium leach testing of intermediate level liquid waste solidified in various cements. The results were obtained on products containing a simulated waste solution whose composition was calculated using the flowsheet of a projected 1460 ton per year fuel reprocessing plant. It is essentially a sodium nitrate solution (850 g/l) containing several heavy metal nitrates, fluoride, phosphate, and organic impurities. All samples were prepared to contain 10% by weight of waste salts. A standard method proposed by IAEA for leach testing of immobilized radioactive waste solids has been widely used in evaluating solidification products; however, meaningful results can only be obtained from it after a leaching time of several months. For this reason it was decided to develop an accelerated leach test to complement the long-term test. The accelerated leach test that has been developed is at elevated temperature, and gives intercomparable results within one day. Using this test, it has been shown that cesium leachabilities from cement products containing a simulated waste typical of fuel reprocessing plants can be reduced by addition of a bentonite. Addition of barium silicate hydrate retards strontium leaching in these cements. Leach rates in tap water and in salt brine are lower than in distilled water and sodium chloride solution. (Auth) (DCW)

Three tables give a) the chemical composition of simulated intermediate-level liquid waste, b) leach coefficients in 80 C distilled water of 6 cements, with and without additives, containing 10% by weight of simulated waste salts, and c) leach coefficients for Cs and Sr in various leachants using the IAEA method at 20 C.

Leaching Rate

Cs 137; Sr 85

WASTES, RADIOACTIVE; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTE PROCESSING; STRONTIUM; CESIUM; IMMOBILIZATION; SOLIDIFICATION; CEMENTS; LEACHING; CHEMISTRY; CHEMICAL PROPERTIES; LABORATORY STUDIES; BENTONITES

<665>

WASTE PRODUCTION

<665>
Bazin, J.E., H.P. Browning, and G.J. McCarthy, Battelle-Pacific Northwest Laboratories, Richland, WA; Battelle Columbus Laboratory, Columbus, OH; Pennsylvania State University, University Park, PA.

Development of Multiburrier Nuclear Waste Forms.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 169-180), 563 pp.(1979)

Research and development activities resulting in the demonstration of the multiburrier concept are described. This concept allows the creation of a composite waste form which enhances inertness through improved thermal stability, mechanical strength, and leachibility by use of metal matrices and coatings. The form produced using this procedure is a solid waste inner core, an impervious coating, and a metal matrix. Already demonstrated as good inner core materials are sintered supercalcine-ceramic and glass marbles. Acceptable coatings are glass and pyrolytic carbon/Al2O3. Four multiburrier concepts are listed in order of increasing technological complexity and inertness: glass marbles encapsulated in a vacuum cast; uncoated sintered supercalcine-ceramic pellets enclosed in vacuum cast; glass coated sintered supercalcine-ceramic pellets encased in vacuum cast; and pyrolytic carbon/Al2O3 CVD coated supercalcine-ceramic encapsulated in gravity-sintered Co. (NDV)

ENCAPSULATION; SOLIDIFICATION; GLASS; CERAMICS; CALCINE; CONTAINERS; LEACHING; MECHANICAL STRENGTH; THERMAL PROPERTIES; WASTE TREATMENT; WASTES, RADIOACTIVE; LABORATORY STUDIES; METHODS

<666>
Savannah River Laboratory, Aiken, SC.

Liquid Waste Evaporation and Acid Recovery.

DPST-APCT-77-1-2; Savannah River Laboratory Quarterly Report - Alternative Fuel Cycle Technologies, (pp. 120-123), 156 pp.(1977, September)

Waste stream categories and waste evaporator systems at Savannah River Laboratory are summarized for design and cost studies. Waste streams from spent fuel reprocessing are classified as High Level Liquid Waste (HLLW) or as Intermediate-Level Liquid Waste (ILLW), depending on fission product, Pu, and U content; Low-Level Liquid Waste (LLLW) is classified as the overheads from ILLW evaporators which require some treatment before release to the environment. For this study, HLLW is arbitrarily subdivided into High Activity Waste (HAW) and Low Activity Waste (LAW). LAW is concentrated and Pu recovered. Concentrated bottoms from the first-stage HAW evaporator and raffinate from the primary recovery column are combined with slurry from the feed clarification step in solvent extraction and stored in HLLW tanks. Overheads from the second-stage HAW evaporator and from the first-stage LAW evaporator are fed into the second-stage LAW evaporator; from here the overheads are transferred to the acid recovery unit. Condensate from the acid recovery unit is either reused as process water or superheated and discharged. ILLW streams are concentrated by one of the following systems: (1) General Purpose Evaporator - concentrates streams associated with spent fuel reprocessing in the as-received acid condition; (2) Service Evaporator - concentrates streams associated with process and process-support functions, with neutralization before evaporation; and (3) Halide Waste Evaporator - concentrates streams containing chloride and fluoride, also with neutralization before evaporation. Concentrated bottoms from all ILLW evaporators are combined and solidified. Overheads from the halide evaporator are re-evaporated, superheated, and discharged to the sandfilter and stack, while those from the other two evaporators are sent to the LLW evaporator. LLW evaporator bottoms are recycled to the service evaporator, the overheads being either superheated and discharged or reused as process water. (LKH)

WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LOW-LEVEL; WASTES, LIQUID; EVAPORATORS; DESIGN; WASTE TREATMENT; RAFFINATES; SLURRIES; EVAPORATION; SOLIDIFICATION; FUEL REPROCESSING

<667>
Savannah River Laboratory, Aiken, SC.

Mathematical Model of Waste Storage Operations.

DPST-75-125-1; Savannah River Laboratory Quarterly Report, Waste Management, January-March, 1975, (pp. 69-75), 84 pp.(1975, January-March)

A computer program for management of a tank farm is presented. The program calculates the month-to-month operations and forecasts the "tank farm" status for any period up to 40 years. At present, input manipulation is required to permit pipeline transfer of suitably dilute wastes from the areas involved and use of new additional tank farms. Further development of the program will automate the use of these operations to facilitate modelling of the current operation of the complex tank farm systems and provide for emptying unserviceable tanks and for efficient modelling of wiped film evaporators. This system would handle both high- and low-level waste. (NDV)

COMPUTER PROGRAMS; EVAPORATORS; TANKS; VOLUME REDUCTION; WASTE MANAGEMENT; WASTE STORAGE; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; THEORETICAL STUDIES

<668>
Savannah River Laboratory, Aiken, SC.

Savannah River Laboratory Quarterly Report, Waste Management, January-March 1975.

DPST-75-125-1; 84 pp. (1975, January-March)

Presented is a quarterly report on the Savannah River Laboratory development program for radwaste management. The topics are divided into two main categories, long-term waste management and interim management of plant waste. Topics under the former heading are: incorporation of waste into glass, bonding mechanisms of concrete

<668> CONT.
waste forms, improved method for determining leachability of radwaste forms, effects of feed composition on Cs removal by ion exchange, residual radionuclides in waste salt solution after Cs removal, use of zeolite for direct removal of Cs from supernate, cesium selective electrode for analysis of high activity waste, solubility of plutonium in waste tank supernate, chemical and thermal characterization of simulated sludge, and volume reduction of plutonium-contaminated soil. Under interim management of plant waste topics include: temperature control of waste tank walls, mathematical model for waste tank cooling, mathematical model of waste storage operation, gamma radiolysis of cellulosic waste, roof vibration of a type 3 waste tank, and in-tank waste solidification with silica. Selected chapters have been abstracted separately. (NDV)

SORPTION; CHEMICAL PROPERTIES; CESIUM; COMPUTER PROGRAMS; CONCRETES; CONTAINER INTEGRITY; FIXATION; GLASS; ION EXCHANGE; LABORATORY STUDIES; LEACHING; MODELS, MATHEMATICAL; PLUTONIUM; SLUDGES; TANKS; TEMPERATURE; VOLUME REDUCTION; WASTE MANAGEMENT; WASTE STORAGE; WASTE TREATMENT; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, LIQUID; WASTES, SOLID; WASTES, RADIOACTIVE; ZEOLITES

<669>
Schulz, W.W., and R.J. Kupfer, Atlantic Richfield Hanford Company, Chemical Technology Laboratory, Richland, WA.

Solidification and Storage of Hanford's High-Level Radioactive Liquid Wastes.

CONF-780386; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, R.B. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 4, (pp. 58-71), 166 pp.(1976)

When the present Hanford Waste Management program is completed in 1980, approximately 180 million liters of solid salt cake, produced by evaporation of aged radioactive liquid waste will be stored in underground steel tanks. Additionally, megacuries of Sr 90 and Cs 137 will have been removed from Hanford waste solutions, converted to SrF2 and CsCl, and stored underwater in doubly encapsulated metallic containers. Several alternative modes for long-term storage/disposal of these high-level wastes are presently being evaluated. For some of these modes conversion of solids to insoluble silicates of low water solubility may be desirable. Laboratory scale studies of both low and high temperature processes for preparation of silicate minerals and glasses are in progress. (Auth) (JMT)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, LIQUID; WASTE STORAGE; WASTE MANAGEMENT; CONTAINERS; GLASS; SILICATES; SOLIDIFICATION; SALT DEPOSITS; WASTE TREATMENT; PHYSICAL PROPERTIES; LEACHING; WASTES, SOLID

<670>
Shapiro, S.A., and G.L. Story, Bechtel Power Corporation, Los Angeles Division, Norwalk, CA.

The Design of Systems for Handling Radioactive Ion Exchange Resin Beads.

Management of Low-Level Radioactive Waste, M.W. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 6, (pp. 563-580), 1214 pp.(1979)

The flow of slurries in pipes is a complex phenomenon. There are little slurry data available on which to base the design of systems handling radioactive ion exchange resin beads and, as a result, the designs vary markedly in operating plants. In late 1975, a survey of plant experience was undertaken to determine what type of problems (if any) existed with current handling systems for high activity spent resin beads. Based on the survey and the then available literature, it was concluded that sufficient slurry data was lacking to properly engineer the design of a radioactive resin bead slurry system. Tests were conducted in conjunction with Robbins & Myers, Hayne Pump Division, to quantify the behavior of resin bead slurries. Based on the test data, the resin-water mixture was classified as a heterogeneous slurry. The tests evaluated the following parameters: resin slurry velocity, pressure drop, bead degradation, and slurry concentration effects. Using visual observations, a velocity of 1.92 m/sec was selected as the lower limit for optimal flow in a 4.1 cm i.d. pipe. This velocity was observed to give full heterogeneous suspension of the resin beads at all concentrations tested. A base design operating slurry concentration of 40% by volume was picked, with 45% by volume as an upper limit. When operating at the recommended minimum slurry velocity and a slurry concentration of 55 wt%, pressure loss in the pipe system increased 40% over that from water alone. Based on these data it is prudent that the engineer apply a sufficient margin over water, at the same velocity, to the pipeline pressure drop in the slurry portions of the system. Degradation of resin beads to fines can contaminate plant systems designed for less radioactive inputs; tests found that about 0.1% by weight per pass of minus 100 mesh fines were generated by the pump, while no degradation was observed from agitation in the drum. The experimental test results presented in this paper provide specific quantitative data necessary to design a resin bead handling system. In conjunction with these test results, the guidance provided in this paper gives the engineer a firm basis on which to make decisions concerning the overall integrated design of a resin handling system. (Auth) (LKN)

Flow Rate

SLURRIES; RESINS; ION EXCHANGE; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTE MANAGEMENT; FLOW RATE; CONCENTRATION; DEGRADATION; PUMPS; DESIGN; EQUIPMENT

<671>
Slansky, C.M., Allied Chemical Corporation, National Reactor Testing Station, Idaho Falls, ID.

High-Level Radioactive Waste Management Progress at the National Reactor Testing Station.

<671>

WASTE PRODUCTION

<671> CONT.
CONF-760946; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, R.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 3, (pp. 31-53), 166 pp.(1976)

Most of the high-level radioactive waste at the National Reactor Testing Station (NRTS) is generated at the Idaho Chemical Processing Plant (ICPP) during the reprocessing of spent nuclear fuel. The high-level radioactive waste is primarily the first-cycle raffinate from the solvent extraction of dissolved fuel solutions. This waste is self-heating in both solid and liquid forms. The high-level radioactive liquid wastes are stored in stainless steel 300,000 gallon tanks located in underground concrete vaults pending their solidification in the Waste Calcining Facility (WCF). The policy at the ICPP is to solidify the high-level radioactive liquid wastes and to store onsite the resulting granular solids in engineered bins where the waste will be in safe interim storage and fully retrievable. In the solid form, the waste is less mobile and the volume is reduced by a factor of 9:1; the cost of solidification and storage is less than building more tanks for liquid wastes. (JMT)

WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; SOLIDIFICATION; WASTES, LIQUID; VOLUME REDUCTION; WASTE STORAGE; WASTE MANAGEMENT; RETRIEVABILITY; RAFFINATES; CONTAINERS; TANKS; PHYSICAL PROPERTIES; CALCINATION; CALCINE; EFFLUENTS

<672>
Steindler, M.J., W.W. Levitz, W.J. Mecham, W.E. Seefeldt, L.E. Trevorrow, I.C. Hinsch, T.J. Gerding, R.J. Sullen, D.S. Webster, and L. Ferris, Argonne National Laboratory, Chemical Engineering Division, Argonne, IL.

Waste Management Program Quarterly Report, July-September 1974.

ANL-8152; 20 pp.(1975, February)

Experimentation in which chemical and mechanical process steps were explored and evaluated for a long-term management scheme for a Zircaloy-based, plutonium-contaminated metal waste stream is presented. Two methods have been selected for initial review: hydrochlorination and reaction with molten zinc chloride. The design work for the hydrochlorination tests has been finished and the apparatus will be built. Preliminary experiments reacting the molten zinc chloride and Zircaloy cutting segments were carried out at temperatures of about 510, 370, and 480 C for 1, 1.5, and 6 hrs, respectively. The reaction was 63, 23, and 69% complete in the respective tests. Apparently temperature is the most significant factor, with the reaction at 510 C being the most vigorous. More tests will be done to monitor the distribution of the minor constituents in this system and determine whether heavy oxide coatings inhibit or interfere with the reaction. Another program is underway to determine methods for decontaminating metals that have been used with plutonium and salvage the metal. Acquisition of the equipment and set-up of an experimental facility is in progress. The facility consists of a glove box with an inert atmosphere feature, arc-melter and a furnace. The furnace will be able to achieve temperatures in excess of 2500 C. A review of a system to concentrate tritium from low-level aqueous waste from fuel reprocessing was also discussed. The system employs an exchange column consisting of concentric tubes with the annulus serving as the reaction chamber. The cost for the system, including capital and operating costs, is about 0.01 mills/kWh of power generated by the nuclear fuel. The system used at Cluj, Rumania is described with equations and diagrams. (NDV)

ACIDS; ALPHA PARTICLES; DECONTAMINATION; EVAPORATORS; EQUIPMENT; ION EXCHANGE; LABORATORY STUDIES; PLUTONIUM; GLOVE BOXES; TEMPERATURE; WASTE MANAGEMENT; WASTE TREATMENT; WASTES, LOW-LEVEL; WASTES, LIQUID; WASTES, SOLID; ZIRCALOYS; TRITIUM; COST BENEFIT ANALYSIS; CHEMICAL PROPERTIES; VOLUME REDUCTION

<673>
Stone, J.A., DuPont Company, Savannah River Laboratory, Aiken, SC.

Studies of Concrete as a Host for Savannah River Plant Radioactive Waste.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 843-852), 563 pp.(1979)

A proposed process for solidifying a type of high-level radwaste in concrete is discussed. Studies were done with simulated sludges in a factorial experiment using 6 types of cement, 3 types of simulated sludge, 1 levels of sludge loading, and 1 levels of water content. The waste/cement ratio increased with increasing sludge content, thus showing the sludges to be hydrophilic. Compressive strengths of concrete-waste castings decreased with increasing sludge content. For strontium the leachability was 10(E-5) to 10(E-3) g/sq cm, and for Pu 239 it was 10(E-8) g/sq cm. The effects of adding cesium-loaded zeolite to cement-sludge mixtures were: the compressive strengths resembled those without the zeolite, and the leachability was about 10(E-3) g/sq cm. Thermal studies were also conducted on the cement mixtures, as were tests with actual waste and the properties were in very good agreement with the simulated sludges. (NDV)

Leaching Rate

SOLIDIFICATION; WASTES, HIGH-LEVEL; ZEOLITES; CEMENTS; SLUDGES; STRENGTH, COMPRESSIVE; LEACHING; CONCRETES; TEMPERATURE; LABORATORY STUDIES

<674>
Stone, J.A., and P.D. d'Entremont, Savannah River Laboratory, Aiken, SC.

Measurement and Control of Cement Set Times in Waste Solidification.

DP-1606; 16 pp.(1976, September)

Set times for mixtures of cement and real or simulated high-level waste sludges were

<676> CONT.

investigated at SRP to determine the optimum formulations for mechanical strength and ease of handling in plant equipment. Decrease the use of excess water to increase set time causes a loss of compressive strength, set retarders were studied. A Plackett-Burman screening test showed that using the recommended amount of the organic set retarder Fozzolith 122-R (0.2 wt% of the dry solids) has no statistically significant effect on set times; set time was controlled mainly by cement type and water content. Tests using faster-setting formulations ("Lumnite" HAC cement with 40% sludge) and retarder contents up to 2.3 wt% of dry solids increased set time, but times were still faster than for HAC neat paste, which sets in 500 min without retarder. Results indicate 1.0-1.5 wt% retarder (5-10 times amount recommended by the manufacturer) would be required with HAC in large-scale operations to insure no setting for the 1-2 hrs necessary for mixing and transfer. Tests on actual SRP waste sludges from Tanks 5, 13, and 15 produced the following results: for Tank 5 sludge at a waste/cement ratio of 0.606 without retarder, and 0.552 with 1.5% retarder, set times were 22 min and 280 min, respectively; for Tank 13 sludge at water/cement ratios of 0.675 without retarder and 0.596 with retarder, set times were 30 and 280 min, respectively; and for Tank 15 sludge at ratios of 0.831 without and 0.775 with retarder, set times were 35 and 290 min, respectively. Proposed future studies will measure compressive strengths of sludge/cement samples prepared with set retarder, radiolytic destruction of retarder, gas production and composition, and control of set times with inorganic retarders. (LKM)

CEMENTS; CONCRETES; LABORATORY STUDIES; SLUDGES; SOLIDIFICATION; IMMOBILIZATION; WASTES, HIGH-LEVEL; ORGANIC COMPOUNDS; WASTE TREATMENT; STRENGTH, MECHANICAL

<675>
Tamura, T., F.S. Brinkley, B.L. Houser, D.G. Jacobs, and C.H. Sealand, Oak Ridge National Laboratory, Oak Ridge, TN.

Sorption and Retention by Clay Minerals: Oak Ridge Studies.

Ground Disposal of Radioactive Wastes, W.J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 54-69), 168 pp. (1961, July)

The ability of clay minerals to sorb and retain radionuclides and the mechanism by which the two processes occur are discussed. Cesium sorption was tested on treated vermiculite and sorption was increased by 85% although the exchange capacity was reduced from 60 meq/100 g to 10 meq/100 g. The sodium treatment converted the c-axis from 14 to 10 A. Montmorillonite sorbed more cesium after treatment, but to get better results heating the clay was necessary. This might have lead to collapse of the montmorillonite structure, impairing its ability to sorbed cesium. However, ion exchange is not a dominant factor in cesium sorption. A sharp decrease in exchange capacity occurs at 700 C for montmorillorite. This and other heating tests indicate the influence of the structural factor is more important than the ion exchange capacity. Further observations show that the collapse of the clay lattice to 10 A increases the cesium sorption. Strontium was found to have an affinity for phosphates and they were most effective at high pH's; i.e. 11. A series of experiments were done to characterize the fixation mechanism. In Arizona bentonite, in kaolinite, and in Wyoming bentonite the entire surface of the clay is accessible to nearly all exchangable cations. Apparently the cesium ion in able to penetrate the open hexagonal network of oxygen atoms and is able to form covalent bonds with the surrounding six oxygens. The illite lattice would allow cesium to be sorbed on the edges. More tests were done with potassium saturated vermiculite. As more potassium is sorbed the exchange capacity decreases. It is noted that vermiculite holds the cesium ion in the lattice truly fixing it. (NDV)

Ion Exchange Capacity

SORPTION; CATION EXCHANGE CAPACITY; FIXATION; ION EXCHANGE; LABORATORY STUDIES; ILLITE; LATTICE; MONTMORILLONITE; KAOLINITE; TEMPERATURE; VERMICULITE; WASTE TREATMENT; WASTES, RADIOACTIVE; CESIUM; SOLIDIFICATION; ENCAPSULATION

<676>
Thomas, H.C., University of North Carolina, Department of Chemistry, Chapel Hill, NC.

The Fundamentals of Ion Exchange Equilibria.

Ground Disposal of Radioactive Wastes, W.J. Kaufman (Ed.), Proceedings of a Conference, Berkeley, CA, August 25-27, 1959. University of California, Berkeley, CA, (pp. 51-53), 168 pp. (1961, July)

The sorption processes are outlined from a thermodynamic point of view. Sorption may be expressed in terms of entropies of mixing and enthalpies of exchange. The discussion centers on the mathematical development of the terms. (NDV)

EQUATIONS; ION EXCHANGE; IONIC PROCESSES; NUMERICAL METHODS; THEORETICAL STUDIES; SORPTION; SOLIDIFICATION; ENCAPSULATION

<677>
Thomas, H.C., University of North Carolina, Chapel Hill, NC.

Mineral Exchange at the University of North Carolina.

TID-7684; Use of Inorganic Exchange Materials for Radioactive Waste Treatment, D.K. Jamison, et al. (Eds.), Proceedings of a Working Meeting, Washington, DC, August 13-14, 1962. U.S. Atomic Energy Commission, Division of Technical Information, Washington, DC, (pp. 13-20), 238 pp. (1963, January)

Two research projects are discussed in this article. One was conducted by Dr. Russell Lewis at the University of North Carolina on a montmorillonite clay from Chambers, Arizona. The clay was cleaned as nearly as possible in order to have a pure clay mineral. Dr. Lewis then studied the sorption properties for three pairs of ions: Cs and Na, Ba and Na, and Cs and Ba. The clay

<677>

WASTE PRODUCTION

<677> CONT.
picked up essentially a constant amount of Ba, approximately 50% of its capacity, and, in every case, the clay picked up a little more Cs than it did Ba. The second experiment was conducted by Mr. Darryl Beatty at the University of North Carolina on the ion-exchange properties of clinoptilolite. 66 grams of the clinoptilolite mineral was leached in a column, with about 32 liters of 1 N BaCl, for the purpose of converting the sodium form and dissolving out any calcium carbonate present. At the end of the elution no trace of calcium could be found. The principal part of this work was to study the Ca-Ba exchange over the whole composition range at two temperatures. The curves for the two temperatures cross near 80% loading for cesium in a region where the mineral is still selective for cesium. Results with barium, strontium, and in particular the hydrogen ion on this material are confusing. When either barium or strontium are sorbed on the mineral, much of the divalent ion appears to be fixed; it is isotopically nonexchangeable with the ion in solution. It is possible that the material still contains enough calcite to obscure the results. (JHT)

MONTMORILLONITE; SORPTION; CLINOPTILOLITE; ION EXCHANGE; EQUILIBRIUM CONSTANT; MINERALS; CLAYS; LABORATORY STUDIES

<678>
Thomas, R.G., E.H. Rex, and C.W. Christenson, Los Alamos Scientific Laboratory, Los Alamos, NM.

Studies on Fixation of Radioactivity in Fired Clays.

TID-7517 (Part 1B); Sanitary Engineering Aspects of the Atomic Energy Industry, Proceedings of a Seminar, Cincinnati, OH, December 6-9, 1955, (pp. 352-367). (1956, October)

Fixation of radwastes in fired clays are studied. Phases used in the study are: filter cake activity fixation, raffinate fixation, mixed fission products, ion exchange capacities, extrusion and mixing, commercial clays, commercial clay combinations, firing and fluxes. Volume reductions using filter cake ranged from 54 to 92%. With raffinate fixation, count retention varied from 99.5975 to 99.9982% using Pu at 8.2 X 10(E-5) gram/liter and Am at 2.7 X 10(E-6) gram/liter. Results with a 1:1:1 combination of waste (800 mCi, mixed fission products), montmorillonite, and bentonite showed only 0.05%, 0.06%, 0.22% leaching in distilled water, tap water, and sea water, respectively. The ion exchange capacities of the local montmorillonite was demonstrated to be 33 meq/100 gram for clay and 30 to 46 meq/100 gram for bentonites, 4 to 6 meq/100 gram for tuff, and meq/100 gram for kaolin. Results from the study show filter cake of an alkaline nature with alpha activity has been permanently fixed and can be disposed by normal burial or dumped at sea. Local clays with raffinate in an acid solution and total alpha count of 126,000 c/m/ml are safe. Distilled water, tap water, simulated sea water, and HNO3 have increasing leaching abilities. The fluxes and firing experimented with have had little success as far as retaining activity is concerned. More work needs to be done. (MDV)

Ion Exchange Capacity

ACIDS; ALPHA PARTICLES; BENTONITES; BETA PARTICLES; CLAYS; CONTAINMENT; FISSION PRODUCTS; RADIATION, GAMMA; ION EXCHANGE CAPACITY; KAOLINITE; LABORATORY STUDIES; LEACHING; MONTMORILLONITE; SAFETY; TUFFS; VOLUME REDUCTION; WASTE TREATMENT; WASTES, LIQUID; WASTES, RADIOACTIVE

<679>
Turcotte, R.P., and J.E. Wald, Battelle-Pacific Northwest Laboratories, Richland, WA.

Devitrification Behavior in a Zinc Borosilicate Nuclear Waste Glass.

PNL-2287; 72 pp. (1978, March)

This report describes experimental studies of thermally induced changes in a simulated nuclear waste glass. The purpose was to define the composition and concentration of all phases formed over a broad range of time (up to 1 yr) and temperature (less than 1200 C). Although the results are specific to a single glass composition (72-68), enough data were obtained to demonstrate that quantitative understanding is possible. Crystallization rates and the approach to equilibrium are shown to follow expected trends based on theoretical/experimental studies of simple glasses. Depending on time and temperature, a steady-state condition is achieved in which a number of crystalline phases coexist with a glass phase which is partially depleted of some elements. Concentrations of the phases increase with decreasing temperature but usually reach only a fraction of their maximum theoretical concentration. Considering the major phase formed (Zn2SiO4), this fractional concentration is about 10% at 900 C and 85% at 700 C, when equilibrium is achieved. The major phases formed and their concentrations (wt%) after one year (at 600 C) are: Zn2SiO4 (14), (Ca,Sr,Ba)MoO4 (6.9), (Ce,Zr,RE 3+)O2-x (5), (Ca,Zn)2(La,Nd,Pr)8 (SiO4)6O2 (3), RuO2 (1.9), and Pd (0.3). Under these unfavorable time/temperature conditions, the glass is about 1/3 crystalline. Because of salt insolubles (RuO2,Pd) the most homogeneous glass will contain about 3% crystalline phases. Crystallization times to reach steady-state concentrations range from a few hours at 900 C to about 1 year at 700 C. No crystallization at 500 C was observed after 1 year. The crystal ingrowth rates follow a reasonable 1/T dependence for Zn2SiO4 and give an activation energy for the process of 82 kcal/mole. Growth rates for the other phases are of the same order of magnitude. Other studies suggest devitrification increases leach rates in this glass by about a factor of 10. Also, the formation of large Zn2SiO4 crystals causes extensive microcracking in the glass matrix. It must also be assumed that actinides may concentrate in phases where rare earth elements appear and that subsequent radiation damage/swelling will cause further microcracking. Therefore, it is desireable to maintain waste glasses in vitreous form. Faster cooling conditions and maintenance of canister centerline temperatures below 375 C are recommended for glass 72-68. It is shown further that crystallization over very long time periods (10(E+6) yr) will likely occur only if ambient temperatures remain above 225 C. (Auth) (LKB)

Appendices provide information on

<679> CONT.
Devitrification products, theoretical considerations of glass crystallization, and quantitative analyses of glass compositions.

Activation Energy

VITRIFICATION; GLASS; FIXATION; DESIGN; CHEMICAL COMPOSITION; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES; THERMAL PROPERTIES; CRYSTALLIZATION; ACTIVATION ENERGY; DEVITRIFICATION; LEACHING; ACTINIDES; RARE EARTHS; WASTE MANAGEMENT; WASTES, RADIOACTIVE

<680>
U.S. Atomic Energy Commission, Technical Information Service Extension, Oak Ridge, TN.

Fixation of Radioactivity in Stable, Solid Media.

TID-7550; Proceedings of a Conference, Baltimore, Maryland, June 19-21, 1957. U.S. Atomic Energy Commission, Oak Ridge, TN; 302 pp.(1958, March)

The developments and progress in treating liquid radwastes so as to fix the radioactivity in a solid form are presented. General topics covered in the meeting were: waste characteristics important in soil fixation; processing alumina fluid calciner into alumina silicates; fixation of fission products; use of ceramic glazes and glasses; leaching studies involving fired clays; separation of cesium and strontium from high-level wastes; disposal of long-lived fission products; high temperature gas cleaning problems; and storage of reactor fuel wastes in salt formations. Several of the discussions have been abstracted separately. (NDV)

SORPTION; CATION EXCHANGE CAPACITY; CERAMICS; CHEMICAL PROPERTIES; CLAYS; DECONTAMINATION; DISTRIBUTION COEFFICIENT; CALCINATION; EVAPORATION; FISSION PRODUCTS; IMMOBILIZATION; LABORATORY STUDIES; LEACHING; GEOLOGY; RADIONUCLIDES; GASES; TEMPERATURE; TANKS; SOILS; WASTES, LIQUID; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL

<681>
U.S. Energy Research and Development Administration, Washington, DC.

The Management and Storage of Commercial Power Reactor Wastes.

ERDA-76-162; 36 pp.(1976)

The processing of spent fuel elements generates four types of radioactive waste. Liquid high-level waste is first to be produced in the solvent extraction cycle. Fission products and small amounts of uranium and transuranic elements comprise this waste. During fuel-element processing solid and liquid non-high-level wastes are generated. One of the major constituents is the protective cladding hulls. The liquids vary widely in chemical and radioactive content. Volatile radioisotopes are evolved during the acid dissolving step and are treated to remove radioactive I 129 before being discharged from the processing facility. Non-high-level wastes are separated into radioactive and nonradioactive components. Evaporation is then used as an initial volume-reduction step. Other processes which may be used are: osmosis, flocculation, precipitation, filtration, and ion exchange. Absorbents are used to immobilize the liquids for transportation and disposal. Vermiculite, silica gelz, Plaster of Paris, and various clays are some of the absorbent materials used. Neutralized evaporator concentrates, sludges, ion-exchange media and incinerator ashes have been immobilized in asphalt or bitumen. For solidified non-high-level wastes the final disposal is burial. Solids, soiled rags, and gloves, are some of the solid wastes and these can be easily compacted. Those wastes which can be incinerated are and the exit gases are filtered to remove harmful levels of radioactivity. Noncombustible solid wastes are treated chemically to remove surface contamination. Mechanical disassembly is another way in which the bulk of radioactive materials can be reduced. Included in the report is a brief introduction to nuclear fission and the fuel cycle. The report also covered fuel cycle option, spent fuel, interim storage, gaseous wastes, storage, transportation, and nuclear waste quantities. The storage section is comprised of a discussion of geologic isolation, the barrier concept, and potential storage sites. (NDV)

ABSORPTION; BURIAL; ACIDS; CEMENTS; CLADDING; DECONTAMINATION; EVAPORATION; EXHAUST GASES; FISSION PRODUCTS; FUEL CYCLES; FUEL REPROCESSING; GEOLOGY; IMMOBILIZATION; INCINERATION; SEPARATION PROCESSES; SLUDGES; WASTE DISPOSAL; WASTE STORAGE; WASTES, GASEOUS; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LIQUID; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; WASTES, SOLID

<682>
Van de Voorde, N., J. Claes, G. Dumont, H. Spriet, and K. Peeters, Centre D'Etude De L'Energie Nucleaire - Studiecentrum voor Kernenergie, Belgium.

Radioactive Waste Processing.

BLG-515/76; Annual Scientific Report 1975, (pp. 9-18 - 9-19), 217 pp.(1976)

Radioactive waste processing by SCK/CEN Radioactive Waste Disposal Department operated by Belgonucleaire in 1975 is summarized. 15,406 cu m of warm liquid waste was released to the intermediate level liquid waste treatment plant; 41,574 kg of dry filtered sludge concentrates were bituminized to 52.9 cu m and conditioned in 302 standard size steel drums or in concrete-lined drums; 260,903 cu m of treated liquid waste (23,905 mCi) was released to the Mothe River, representing 84% of the total weighted activity release permitted to the river annually. 482 cu m of solid low-level combustible waste was incinerated to 17.4 cu m of ash, bituminized, and conditioned in concrete-lined drums; 824 cu m of incombustible and alpha-contaminated solid wastes were baled, bituminized, and conditioned (reduction factor 1.85); 827 cu m of non-combustible, incompressible waste was bituminized and conditioned in steel drums; 8 cu m of media and high-level waste was stored at Interim Storage Site I; total vented stored equals 210 Ci alpha, beta, gamma, 593 Ci tritium, 216.6 g Pu, and 82.3 kg U. 2001 metric tons of insolubilized

<682> CONT.
solid waste with a gross alpha, beta, gamma activity of 266 Ci and 3,025 Ci tritium was dumped at a selected site 4500 m deep in the Atlantic Ocean. (LKH)

Waste Volume

H 3; Pu; U

WASTES, LIQUID; WASTES, RADIOACTIVE; WASTE TREATMENT; WASTE MANAGEMENT; WASTES, SOLID; BITUMINIZATION; SEA DISPOSAL; INCINERATION; IMMOBILIZATION; TRITIUM; URANIUM; PLUTONIUM; CONCRETES; PLANTS, WASTE TREATMENT; WASTE VOLUME

<683>
Wallace, R.M., W.H. Hale, P.P. Bradley, H.L. Bell, J.A. Kelley, J.A. Stone, and G.M. Thompson, Savannah River Laboratory, Aiken, SC.

Solid Forms for Savannah River Plant Radioactive Wastes.

CONF-740346; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, M.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 2, (pp. 5-30), 166 pp.(1976)

The radioactive wastes from plant operations at the Savannah River Laboratory are stored as liquids or slurries in double-walled carbon steel tanks. Until recently, the only alternative under serious consideration was storage as a liquid in a deep mined cavern under the plantsite, but in November of 1972, the AEC deferred further study of that project in favor of an investigation of the solidification of SRL waste and storage in a Retrievable Surface Storage facility. Methods are now being developed to immobilize these wastes in solid forms such as cement, asphalt, or glass. Cs 137 and Sr 90 are the major biological hazards and heat producers in the alkaline wastes produced at SRL. In the conceptual processes studied, Cs 137 removed from alkaline supernates, together with insoluble sludges that contain Sr 90, will be incorporated into solid forms of high integrity and low volume suitable for storage in a retrievable surface storage facility for about 100 years, and for eventual shipment to an offsite repository. Mineralization of Cs 137, or its fixation on zeolite prior to incorporation into solid forms, is also being studied. Economic analyses to reduce costs and fault-tree analyses to minimize risks are being conducted. Methods are being studied for removal of sludge from (and final decontamination of) waste tanks. (JHT)

WASTE MANAGEMENT; DESIGN; STORAGE; SOLIDIFICATION; CEMENTS; GLASS; ASPHALT; WASTES, RADIOACTIVE; WASTE STORAGE; TANKS; CHEMICAL PROPERTIES; SLUDGES; ION EXCHANGE; FIXATION; RETRIEVABILITY; COST BENEFIT ANALYSIS

<684>
Walton, R.D., Jr., U.S. Energy Research and Development Administration, Washington, DC.

Radioactive Waste Management Development in Europe.

CONF-740346; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, M.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 11, (pp. 158-160), 166 pp.(1976)

Germany, England, and France are vigorously developing technology and methodology for incorporating high-level radioactive waste into silicate glass. Germany is concentrating on a spray calcination vitrification system. England has selected a rising level glass process in which evaporation, calcination, and borosilicate glass vitrification all take place in a heated pot. France is operating a small-scale batch pot calcination-batch vitrification system and developing a new continuous system using a rotary calciner and melter with batchwise draw-off. (Auth)

WASTE MANAGEMENT; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; GLASS; CALCINATION; REVIEWS; WASTE TREATMENT

<685>
Weeren, H.O., and J.J. Perona, Oak Ridge National Laboratory, Oak Ridge, TN.

A Preliminary Engineering and Economic Analysis of the Fixation of High-Level Radioactive Wastes in Concrete.

ORNL/TM-6863; 59 pp.(1979, July)

This study of the engineering and economic feasibility of using concrete for the fixation of defense or commercial high-level wastes was made to (1) determine the inherent problems of the process, (2) determine areas in which more information is needed, and (3) make some preliminary cost comparisons as an indication of the overall economic practicability of the process. The study was based on a waste fixation facility that would serve as a reprocessing plant with a capacity of 5 metric tons of uranium per day. Postirradiation cooling times of 3 to 10 years prior to waste solidification were assumed. The waste solution would be concentrated, denitrated, mixed with cement, and cast under pressure in cylindrical canisters similar to those envisioned for a glass facility. The solidified waste grout would be vented to allow free water to escape and then sealed. The filled canister would be shipped to a geologic repository for permanent storage. Concrete fixation of high-level wastes would require that the waste grout be cast at temperatures and pressures high enough for the HOTRAP (formed under elevated temperature and pressure) phenomena to have a significant effect on product properties. Operating costs for a waste fixation facility were estimated under several conditions. The largest part of the operating cost would be repository charges (70-90% of the annual cost). Because this cost is so large, the method selected for the allocation of repository charges will affect the comparative costs much more than will process variables. In comparison with a glass fixation facility, operating costs for the concrete facility are about 12% higher; however, the capital cost of a glass facility would be 60-80% more. Therefore, the overall annual operating cost (amortized capital plus direct operating cost) would be somewhat lower for the concrete facility. The principal conclusion of this study is that

<685> CONT.
concrete could be an alternative to glass as a matrix for fixation of high heat-generation wastes. Since the concrete process is not a high-temperature process, it would not have the consequent operating problems. Conclusive data on the long-term stability of the concrete waste form are not available; the conclusions of this report are based on process and economic considerations only. (Auth) (LKM)

WASTE PROCESSING; CONCRETES; FIXATION; WASTES, HIGH-LEVEL; COSTS; COST ESTIMATES; COST BENEFIT ANALYSIS; ECONOMICS; CANISTERS; CONTAINERS; VITRIFICATION; GLASS; THEORETICAL STUDIES

<686>
Weisenberger, S., Kernforschungszentrum, Karlsruhe, German Federal Republic.

Non-Radioactive Operation Experience with a Joule Heated Ceramic Melter for Vitrification of High-Level Liquid Waste.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY (pp. 51-56), 563 pp. (1979)

For vitrification of high-level liquid waste (HLLW) a Joule heated ceramic melter system has been developed. Essential features of this system are fully continuous operation and especially the feeding of HLLW directly into the melter. Two direct feeding methods have been thoroughly investigated. The first was feeding HLLW through an atomizing nozzle from the top of the melter; the second one was using a tube of 6 mm in diameter and transferring the liquid waste immediately to the surface of the molten glass. The subjects of this paper are the operating principles of the melter, and the experience gained by continuous operation of the pilot-scaled melter system at Karlsruhe Nuclear center. (Auth) (Complete Text)

SOLIDIFICATION; VITRIFICATION; WASTES, HIGH-LEVEL; WASTES, LIQUID; GLASS; CERAMICS; METHODS; LABORATORY STUDIES

<687>
Willing, H.W., and D.W. Rhodes, Allied Chemical Corporation, National Reactor Testing Station, Idaho Falls, ID.

Removal of Cesium and Strontium from Fuel Storage Basin Water.

CONF-760386; Advances in Chemistry Series 153; High-Level Radioactive Waste Management, M.H. Campbell (Ed.), Proceedings of the 167th Meeting of the American Chemical Society, Los Angeles, CA, April 1-2, 1974. American Chemical Society, Washington, DC, Ch. 10, (pp. 138-151), 166 pp. (1976)

Spent fuel from nuclear reactors is stored underwater at the Idaho Chemical Processing Plant for cooling and shielding before processing. The fuel storage basin water becomes contaminated with fission products, primarily cesium 137 and strontium 90, from fuel elements and from cut pieces of fuel and miscellaneous scrap contained in cans, which are vented to release gases. Laboratory research and plant-scale tests for ion-exchange materials to remove cesium 137 and strontium 90 from the contaminated waters has been done. Cesium 137 is removed by a zeolitic ion-exchange material; strontium 90 is removed by an organic ion-exchange resin. Operational experience with plant-size ion-exchange columns indicate that both cesium 137 and strontium 90 are removed effectively by ion exchange. (Auth) (JMT)

FUEL ELEMENTS; WASTE STORAGE; CESIUM; STRONTIUM; ION EXCHANGE; RESINS; WATER; ZEOLITES; LABORATORY STUDIES; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; WASTES, RADIOACTIVE; PHYSICAL PROPERTIES; CHEMICAL PROPERTIES; DECONTAMINATION

<688>
Wiley, J.R., Savannah River Laboratory, Aiken, SC.

A Study of Methods for Removing Strontium, Plutonium, and Ruthenium from Savannah River Plant Waste Supernate.

DP-1408; 18 pp. (1976, June)

Methods of removing strontium, plutonium, and ruthenium from alkaline waste supernate are discussed in this report. Duolite ARC-359 resin sorbed plutonium and cesium from simulated supernate. Strontium was removed from the supernate by sorption on a chelating resin Chelex 100 or by precipitation as $Sr_3(PO_4)_2$. None of the methods tested were adequate for plant-scale removal of ruthenium. (JMT)

PLUTONIUM; STRONTIUM; RUTHENIUM; RESINS; ION EXCHANGE; CESIUM; SORPTION; VOLUME REDUCTION; WASTE TREATMENT; WASTES, RADIOACTIVE; WASTES, HIGH-LEVEL; WASTES, LIQUID

<689>
Zemlyanukhin, V.I., Yu.V. Kuznetsov, L.N. Lazarev, N.I. Lyubtsev, and E.A. Shashukov, V.G. Khlopin Radium Institute, Leningrad, USSR.

Clay-Phosphate Ceramics and Vitromets: Alternatives to Monolithic High Level Waste Products.

CONF-781121; Scientific Basis for Nuclear Waste Management, Science Underlying Radioactive Waste Management, Vol. 1, G.J. McCarthy (Ed.), et al., Proceedings of a Symposium, Boston, MA, November 28-December 1, 1978. Plenum Press, New York, NY, (pp. 195-200), 563 pp. (1979)

Alternatives to vitrification of liquid high-level wastes (HLW) were investigated in order to find a more stable waste form than glass monoliths to produce a waste form at a lower cost, and to meet more modest requirements for safe transportation of wastes to burial sites. Efforts were focused on HLW incorporation into ceramic, cermet and vitromet products. Tests were done to improve production methods with regard to radiochemical specificity and to define the properties of products with specific reference to their subsequent use. Feed materials were Cambrian clay and kaolin. After mixing with water (up to 15 wt%), these minerals form a plastic mould paste. The

<689> CONT.

plastic properties are retained on addition of either 50 wt% of a simulated oxide-hydroxide calcine, or a phosphate product resulting from the interaction of the metal nitrates with orthophosphoric acid. For a study of the properties of various products, pellets were pressed at 1000 kg/sq cm at room temperature. The properties of primary (feed), intermediate and final products prepared according to different composition, temperature, drying and annealing conditions were studied. Sodium leachability from the products in water, and strength and open porosity were determined. The products were also studied by thermogravimetric and X-ray phase analysis. With increasing temperature, both the open porosity and the sodium leachability decreased. During the first few days, the leachability of sodium from the pellets treated at 850-900 C was of the order of 10(E-5) g/sq cm d. For both the Cambrian clay and the kaolin products treated at 800-850 C, hardness was found to be only 1.5 times lower than the hardness of tempered steel. Also investigated was a vitromet as a potential matrix for radioactive wastes. Vitromets consist of glass granules encased in a high heat-conducting and stable metallic matrix. In such a product, it is advisable to make the waste loading of the glass granules as large as possible in order to reduce the volume of the products to be disposed of. For the formation of the most concentrated melts containing of all of the fission products and other substances present in HLW, phosphate glass is of interest. In such a glass, up to 60 wt% of a simulated calcine can be incorporated. The resulting products were homogeneous and had high non-crystallinity as confirmed by X-ray analysis, IR spectroscopy, and DTA. However, when the concentration of Na_2O exceeded 22 wt% the formation of a crystalline phase was observed. (RAF)

Porosity; Leaching Rate

WASTE FORM; CERAMICS; WASTES, HIGH-LEVEL; WASTES, LIQUID; CLAYS; KAOLIN; CALCINE; PHOSPHATES; LABORATORY STUDIES; LEACHING; TEMPERATURE; POROSITY; GLASS; METALS; WASTES, RADIOACTIVE; WASTE TREATMENT

<690>
Alexander, C.W., C.W. Kee, R.G. Croff, and J.O. Blomeke, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN.

Projections of Spent Fuel to be Discharged by the United States Nuclear Power Industry.

CONF/TR-6008; 83 pp.(1977, October)

Calculated properties of spent fuel projected to be discharged and accumulated by the U.S. nuclear power industry through the year 2031 are presented. The projections are based on installed nuclear capacities of 386 and 543 Gw(e) in the year 2000 and 2030, respectively. They include compilations of the grams of the elements, curies of radioactivity, thermal decay power, photon and neutron emission rates, and radiotoxicities of the assemblies that are accumulated at a Spent Unprocessed Fuel Facility (SUFFF), allowing for delays of 5 and 10 years before shipment to SUFFF. The nuclear power generating capacities used in this report are based on the April 1977 "Low Growth Case" of the Energy Research and Development Administration's (ERDA) Office of Planning, Analysis, and Evaluation through 2000 AD, and on a Battelle-Pacific Northwest Laboratories projection thereafter. The "Low Growth Case" assumes indefinite deferral of reprocessing; the Battelle forecast assumes a nuclear industry consumption of 3.7 million short tons of U3O8 in its operating lifetime with an enrichment tails assay of 0.1% U 235. In addition, it assumes the following reactor capacity factors: operation year 1 = 82%, years 2-3 = 65%, years 4-15 = 70%, and years 16-40 = declines 2%/yr to 40%. This is equivalent to a 40-yr lifetime capacity factor of 56%. Characteristics of reference uranium-enriched light water reactors (LWR) were taken from boiling water reactor (BWR) and pressurized water reactor (PWR) standard safety analysis reports and are based on 1250 Mw of electrical generation, burnups of 27,500 and 33,000 Mwd/metric ton for BWRs and PWRs, respectively, and charge-discharge rates based on an 80% reactor capacity factor. Isotopic buildup and depletion in the two types of reactors were calculated with a modified version of the ORIGEN code, designated ORIGEN2. Approximately 200 of the cross sections used in ORIGEN2 resulted from BWR and PWR reactor physics calculations using the XSDRNPM and CITATION codes. These calculations considered the buildup and depletion of 38 actinide isotopes between Th 232 and Es 253. Eighteen of these cross sections vary with burnup during the ORIGEN2 irradiation calculations. Quantities of spent fuel discharged annually were derived from these data using the KWIKPLAN code. ORIGEN2 was then used to calculate the properties of the spent fuel accumulated through 2031. In these projections, allowance was made for 2382 metric tons of uranium reported by ERDA to be in storage as commercial spent fuel at the end of 1976. Three different projections of spent fuel inventories were made. One was concerned with the total spent fuel inventory, as generated in the U.S. through 2031; the other two were concerned with spent fuel inventories at SUFFF. The Nuclear Assurance Corporation (NAC) has made projections of spent fuel to be discharged through 1986. These projections differ by up to 50% from the projections in this report. This is due first to the fact that SUFFF projections are based on the more conservative ERDA "Low Growth Case". Second, the SUFFF projections are based on reactor capacity factors, while NAC projections use individual utility discharge strategies. Finally, the SUFFF projections use reactor models to predict the amount of fuel discharged, which cannot reflect the characteristics of all presently operating reactors exactly. (Auth)(LKN)

Volume projections for many radionuclides through the year 2031 are provided in extensive tables.

WASTE VOLUME; PREDICTIONS; COMPUTER CODES; SPENT FUELS; NUCLEAR POWER; WASTE MANAGEMENT; WASTES, COMMERCIAL; WASTES, RADIOACTIVE; REACTORS, BOILING WATER; REACTORS, LIGHT WATER; REACTORS, PRESSURIZED WATER; ACTINIDES; BURNUP; DISCHARGE; URANIUM; MODELS, MATHEMATICAL; DISPOSAL FACILITIES; THEORETICAL STUDIES

<691>
Bell, M.J., and J.T. Collins, U.S. Nuclear Regulatory Commission, Division of Site Safety and Environmental Analysis, Effluent Treatment Systems Branch, Washington, DC.

Sources of Radioactive Waste from Light-Water Reactors and Physical and Chemical Properties.

Management of Low-Level Radioactive Waste, M.W. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 2, (pp. 79-88), 1214 pp.(1979)

This paper briefly describes the general physical and chemical properties of waste streams in light-water reactors (LWRs), describes the principal mechanisms for release and the release pathways to the environment, discusses the calculation of liquid and gaseous source terms using one of the available models, and compares these calculated releases with observed releases from operating LWRs. The computerized mathematical model used is the GALE Code which is the Nuclear Regulatory Commission (NRC) staff's model for calculating source terms for effluents from LWRs. In addition, the paper discusses programs currently being conducted at operating reactors by the NRC, Electric Power Research Institute, and various utilities to better define the characteristics of waste streams and the performance of radwaste process equipment. The critical isotope released in gaseous effluents from LWRs is I 131. Calculated releases for 6 boiling water reactors (BWRs) considered were from 0.46 to 0.69 Ci/yr/reactor of I 131, and 9200 to 87,000 Ci/yr/reactor of noble gases, while actual releases reported in 1976 were 2.2 X 10(E-3)-0.53 Ci/yr/reactor of I 131 and 2900-100,000 Ci/yr/reactor of noble gases. Calculated releases from 4 pressurized water reactors (PWRs) were 0.013-0.083 Ci/yr/reactor of I 131 and 2900-9700 Ci/yr/reactor of noble gases. Actual releases in 1976 were 1.6 X 10(E-3)-0.04 Ci/yr/reactor of I 131 and 1300-6300 Ci/yr/reactor of noble gases. Comparison of calculated and actual releases of radioactive liquid effluents were 0.15-4.4 Ci/yr/reactor versus 0.22-6.0 Ci/yr/reactor (1976) for 6 BWRs studied, and 0.24-1.0 Ci/yr/reactor versus 0.44-2.8 Ci/yr/reactor (1976) for 4 PWRs studied. The calculated values show reasonable agreement with the releases reported by the licensees. Thus, these models reasonably characterize the expected releases over the life of the plant. (Auth)(LKN)

<691>

WASTE TREATMENT

<691> CONT.
MODELS; MODELS, MATHEMATICAL; IODINE 131; WASTES, GASEOUS; WASTES, LIQUID; EFFLUENTS, RADIOAC; EFFLUENTS, LIQUID; INERT GASES; PREDICTIONS; WASTES, RADIOACTIVE; WASTES, LOW-LEVEL; WASTE VOLUME; REACTORS, PRESSURIZED WATER; REACTORS, LIGHT WATER; REACTORS, BOILING WATER; NUCLEAR FACILITIES; COMPUTER CODES; COMPUTER PROGRAMS; WASTE MANAGEMENT; CHEMICAL PROPERTIES; PHYSICAL PROPERTIES

<692>
Chapin, J.A., and R.E. Hine, EG and G Idaho, Incorporated, Idaho Falls, ID.

Contaminated Reactor Facilities.

IDEE-1250 (Volume 2); Decontamination and Decommissioning, Long Range Plan, Idaho National Engineering Laboratory, C.R. Tolman (Ed.), Appendix A, (pp. A-1 - A-90), 319 pp. (1978, June)

The operational and nonoperational facilities of the Idaho National Engineering Laboratory (INEL) Test Reactor Area (TRA) are described as to type and function of facility. Wastes generated are indicated and treatment methods are outlined. Control of contaminated liquid waste at the TRA involves a series of controls at each source. Additional controls are also located at the retention basin where the waste from various sources is combined for ultimate disposal in the leaching ponds. Reactors discussed are: Materials Testing (MTR), Engineering Test (ETR), Advanced Test (ATR), and four low power reactors--the Engineering Test Reactor Critical Facility, Advanced Test Reactor Critical Facility, and two Advanced Reactivity Measurement Facilities. The Power Burst and Loss-of-Fluid Test Facilities are also discussed. Under the nonoperational facilities are SPERT I-IV, Materials Testing Reactor, BORAX-V, and Organic Moderated Reactor Experiment. Each primary reactor area (ETR, MTR, and ATR) and the support facilities area have a specific liquid waste disposal system; all areas use common discharge points. (NDV) (CSF)

Schematic diagrams of the various facilities are included.

NUCLEAR FACILITIES; PLANTS, WASTE TREATMENT; REACTORS; PITS; TANKS; WASTE DISPOSAL; WASTE STORAGE; WASTE TREATMENT; WASTES, LIQUID; DECONTAMINATION; WASTES, LOW-LEVEL; WASTES, HIGH-LEVEL; SAMPLES; CONTAMINATION; WASTE VOLUME; TEMPERATURE

<693>
Chapin, J.A., R.E. Hine, and C.R. (Ed.) Tolman, EG and G Idaho, Inc., Idaho Falls, ID.

Decontamination and Decommissioning, Long Range Plan, Idaho National Engineering Laboratory.

IDEE-1250 (Volume 2); 319 pp. (1978, June)

A detailed description of the various facilities for decontamination and burial is provided. Included in the discussion, which is divided into appendices, are: contaminated reactor facilities, reprocessing facilities, Idaho Chemical Processing Plant, description of laboratory facilities, burial grounds, description of contaminated miscellaneous equipment and areas, and contaminated inventory at the Idaho National Engineering Laboratory. All appendices were abstracted separately. (NDV)

Several diagrams are given in the appendices.

BURIAL; CALCINATION; CONTAMINATION; DECONTAMINATION; DISPOSAL SITE; EVAPORATION; ION EXCHANGE; MONITORING; NUCLEAR FACILITIES; PITS; PLANTS, WASTE TREATMENT; REPROCESSING; REPOSITORY; RETRIEVABILITY; SOLIDIFICATION; TANKS; STORAGE, ABOVEGROUND; STORAGE, GEOLOGIC; URANIUM; WASTE MANAGEMENT; WASTE STORAGE; WASTE TREATMENT; WASTE VOLUME; WASTES, HIGH-LEVEL; WASTES, INTERMEDIATE-LEVEL; WASTES, LOW-LEVEL; WASTES, SOLID; WASTES, TRANSURANIC; DECOMMISSIONING

<694>
Davis, W., Jr., Oak Ridge National Laboratory, Oak Ridge, TN.

Carbon 14 Production in Nuclear Reactors.

Management of Low-Level Radioactive Waste, A.M. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 2, (pp. 151-191), 1214 pp. (1979)

Quantities of C 14 that may be formed in the fuel and core structural materials of light-water cooled reactors (LWRs), in high-temperature gas-cooled reactors (HTGRs), and in liquid metal cooled fast breeder reactors (LMFBRs) have been calculated by use of the ORIGEN computer code. Information supplied by five LWR-fuel manufacturers pertaining to nitride nitrogen and gaseous nitrogen in their fuels and fuel rod void spaces was used in these calculations. Average nitride nitrogen values range from 3 to 50 ppm by weight in LWR fuels, whereas gaseous nitrogen in one case is equivalent to an additional 10-16 ppm. Nitride nitrogen concentrations in fast-flux test facility (FFTF) fuels are 10 to 20 ppm. The principal reactions that produce C 14 involve N 14, O 17, and (in the HTGR) C 13. Reference reactor burnups are 27,500 MWd per metric ton of uranium for boiling water reactors (BWRs), 33,000 MWd for pressurized water reactors (PWRs), about 95,000 MWd per metric ton of heavy metal (MTHM) for HTGRs, and 28,800 MWd/MTHM for an LMFBR with nuclear parameters that pertain to the Clinch River Breeder (CRBR). Nitride nitrogen, at a median concentration of 25 ppm, contributes 27, 19, and 6 Ci of C 14/GW(e)-yr to BWR, PWR, and FFTF fuels, respectively. The contribution of O 17 in BWR and PWR fuels is 8.0 and 3.5 Ci of C 14/GW(e)-yr, respectively, but it is less than 0.2 Ci/GW(e)-yr in blended LMFBR fuel. In the HTGR fuel particles (UC2 or ThO2), 10 Ci of C 14/GW(e)-yr will be formed from 25 ppm of nitrogen, whereas O 17 in the ThO2 will contribute an additional 2 Ci/GW(e)-yr. All C 14 contained in the fuels may be released in a gas mixture (CO2, CO, CH4, etc.) during fuel dissolution at the fuel reprocessing plants. However, some small fraction may remain in aqueous raffinates and will not be released until these are converted to solids. The gases would be released from the plant unless special equipment is installed to retain the C 14-bearing gases. Cladding metals and other core hardware will contain significant quantities of C 14. Very little of this will be released from BWR, PWR, and LMFBR hardware

<696> CONT.

at reprocessing plants; instead, the contained C 14, 30-70 Ci/GW(e)-yr for LWRs and about 13 Ci/GW(e)-yr for a CHBR, will remain within the metal, which will be retained on site or in a federal repository. The only core structural material of HTGRs will be graphite, which will contain 37-190 Ci of C 14/GW(e)-yr, exclusive of that in the fuel particles, if the graphite (fuel block and reflector block) initially contains 0-30 ppm of nitrogen. All of this is available for release at a fuel reprocessing plant if the graphite is burned to release the fuel particles for further reprocessing. (Auth)

CARBON-14; REACTORS, LIGHT WATER; REACTORS, HIGH-TEMPERATURE GAS-COOLED; REACTORS, LIQUID METAL FAST BREEDER; REACTORS; COMPUTER CODES; FUELS; NITROGEN-14; OXYGEN 17; BURNUP; FUEL RODS; GASES; EFFLUENTS, AIRBORNE; FUEL REPROCESSING; GRAPHITES; HAZARD ANALYSIS; PREDICTIONS; THEORETICAL STUDIES; WASTE MANAGEMENT; WASTE TREATMENT

<695>

Davis, W., Jr., R.E. Blanco, D.C. Finney, G.S. Hill, R.E. Moore, and J.P. Witherspoon, Oak Ridge National Laboratory, Chemical Technology Division, Oak Ridge, TN; Oak Ridge National Laboratory, Environmental Sciences Division, Oak Ridge, TN.

Miscellaneous Radioactive and Nonradioactive Wastes.

CONF/NUREG/IN-4; Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing Uranium 233 and Thorium, Section 5.0, (pp. 68-67), 182 pp. (1976, May)

The operation of an high-temperature gas-cooled reactor nuclear fuel reprocessing plant will generate various radioactive and nonradioactive wastes, such as reflector blocks, fuel element parts, discarded equipment, laboratory waste, sanitary waste, process cooling water, combustion products, etc. In addition, significant volumes of high- and low-activity liquid wastes are accumulated in storage tanks. Stainless steel cylinders filled with solid high-activity wastes will be produced at the plant since it will have facilities for solidification of the liquid waste. Estimates of the amounts of these wastes and a discussion of methods for their management at the model reprocessing plant that handles 850 metric tons of heavy metal (U + Th charged to the reactors) per year are presented in this section. (Auth) (PAT)

WASTES, SOLID; GRAPHITES; EQUIPMENT; REACTORS, HIGH-TEMPERATURE GAS-COOLED; REPROCESSING; WASTES, LIQUID; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; BURIAL; WASTE MANAGEMENT; IODIDES; WASTE VOLUME; THORIUM; URANIUM 233; WASTES, COMMERCIAL; WASTES, ORGANIC; GASES; NITROGEN OXIDES; THEORETICAL STUDIES

<696>

Dyroff, H., and F.K. Fleischmann, NUKEM GmbH, Hanau, German Federal Republic.

Radioactive Wastes in the Federal Republic of Germany: Accumulation of Radioactive Wastes in the Federal Republic of Germany.

NEDO-tr-306 (2nd Ed.); 75 pp. (1976, June)

A study is presented on the amount of radwastes to be produced in Germany in the future. This is part of an effort to prorate the ability of facilities to handle the waste volumes. An average growth rate of installed nuclear power plant capacity from 1975 to 2000 was determined using the build-up programs and the most recent data. Data for other users of radioactive materials was derived by special programs so an overall volume could be ascertained. From this work a waste catalog was compiled. The analysis of waste volume in Germany from 1975 to 1990 indicates that nuclear power plants generate the largest waste volume. The specific waste types established in this report are used as initial material for inventory and evaluation of available processing technology and current research and development works for radioactive wastes. A removal model is selected and action program for construction of this system is proposed with consideration of safety and administration regulations. (NDV)

ALPHA PARTICLES; CLASSIFICATION; FUEL REPROCESSING; HAZARD ANALYSIS; MODELS; NUCLEAR FACILITIES; WASTE MANAGEMENT; WASTE TREATMENT; WASTE VOLUME; WASTES, RADIOACTIVE

<697>

Gore, B.F., J.A. Kaser, and T.J. Kabele, Battelle-Pacific Northwest Laboratories, Richland, WA.

Fusion Fuel Cycle Solid Radioactive Wastes.

PNL-2719; 29 pp. (1978, June)

Eight conceptual deuterium-tritium fueled fusion power plant designs have been analyzed to identify waste sources, materials, and quantities. All plant designs include the entire D-T fuel cycle within each plant. The choice of materials within the reactor blanket is the controlling factor in fusion reactor waste generation. Choice of blanket structural material controls radiation damage rates and activation products created; chemical compatibility of blanket structure with fertile material and coolant controls corrosion rates and limits the blanket operating temperatures. The 8 conceptual fusion power plant designs studied in this report are the UWMAK-1, BNL Blanket, UWMAK-2, Princeton, and ORNL Tokamak systems, the LLL Mirror, the LASL Theta Pinch, and the LASL Inertial systems. Primary fusion reactor radwastes identified from design documents were replaced blanket structures, corrosion products and spent getters, containment air tritium absorbents, bred tritium recovery materials, fuel recycle purification wastes, vacuum system materials, miscellaneous failed equipment, and decontamination and laundry wastes. Estimated annual wastes quantities from replacement of structures in the blanket in t/GWe were: stainless steel, 110-490 for the UWMAK and Princeton Tokamak systems, 192 for the LLL Mirror, and 90 for the LASL Laser; niobium, 27 for CRSL Tokamak, 19 for LASL Theta Pinch; aluminum, 144 for BNL Tokamak; and other materials, 180-212 for the BNL, ORNL and UWMAK-2 Tokamak systems, 66 for LLL Mirror, and 30 for LASL Theta Pinch.

<697>

WASTE TREATMENT

<697> CONT.

Thirty-two metric tons of LWR spent fuel rods would be removed per GWe. Estimated volumes (in cu m/GWe) of solid radwaste generated by the conceptual designs analyzed are: blanket replacement, 20-160; molecular sieve material, 30; getters and recovered products, 3-60; failed equipment, 15-35; and decontamination wastes, 80-220. Total waste generated would be 150-600 cu m/GWe, as compared with 500-1300 cu m/GWe estimated for the LMFBR fuel cycle. The estimates for decontamination wastes are very uncertain; also, current research efforts to increase fusion reactor economy should considerably reduce blanket replacement wastes. (Auth) (LKH)

REACTORS, FUSION; THEORETICAL STUDIES; WASTE MANAGEMENT; WASTE VOLUME; WASTES, RADIOACTIVE; WASTES, LIQUID; TRITIUM; WASTES, SOLID; DECONTAMINATION; EQUIPMENT; FILTERS; REACTOR BLANKET; STAINLESS STEELS

<698>

Granquist, R.P., and R.E. Toulinson, General Electric Company, Hanford Atomic Products Operation, Richland, WA.

Estimate Magnitude of Waste Disposal Problems.

TID-7517 (Part 1B); Sanitary Engineering Aspects of the Atomic Energy Industry, Proceedings of a Seminar, Cincinnati, OH, December 6-9, 1955, (pp. 400-405). (1956, October)

Disposal of fuel cycle radwastes are discussed. Also considered is the magnitude of the problem and some criteria for problem evaluation. It is assumed that economics will require the irradiated fuel element be reprocessed for the recovery of fissile materials. For purposes of this discussion, a unit block of power of 100 million kilowatt of heat energy, representing perhaps 10 to 20 million kilowatts of electrical energy is considered. Fission releases about 200 million electron volts of energy per atom fissioned. More than 99% of this energy is available as heat from the reactor on a continuing basis, while the rest is the decay energy as beta and gamma emissions. Assuming a constant rate of heat production at 100 million kilowatts and a fuel irradiation equivalent to 1 to 2 megawatts/kilogram of U 235 for 250 days, then this fission product decay would release heat at a rate of about 50 to 100 million BTU/hr. This heat will be dissipated from an aqueous solution in a tank, cast in concrete at the bottom of the ocean, or fused into a glass in the middle of a desert. Assuming 100 million kilowatts of heat energy was produced by the irradiation of natural or slightly enriched uranium to about 2500 megawatt-days, 20,000 gallons of highly radioactive liquid waste would be produced per day. These wastes might be self-concentrated by a factor of three on a continuing operating basis.
Self-concentration would occur when the solution would evaporate by the heat evolved from the fission product decay. The self-concentrated wastes would accumulate at about 2 million gallons/year and would be expected to boil for about 50 years. Storage of non-boiling wastes is in 500,000 to 1 million gallon tanks at about 20 to 25 cents/gallon. Storage of boiling wastes costs about 40 to 50 cents/gallon. The estimated costs of storing wastes from the reprocessing of fuel elements are 0.01 to 0.05 mils/kilowatt-hr of electrical power produced. This cost amounts to less than 1% of the total cost of producing atomic power. At present, semi-permanent storage in tanks is economically viable. An alternative method would be to extract Cs 137 and Sr 90 and store the rest of the wastes in a soil with sufficient ion exchange capacity to retain them. Other methods under consideration are: calcination to a salt or oxide, preparation of a concrete or a glass and disposal in tanks or caverns, or burial in a desert or at sea. Those all have drawbacks. The ideal solution would be to develop a market for these wastes. (NDV)

Waste Volume

BETA PARTICLES; CALCINATION; CONCENTRATION FACTORS; CONCRETES; CONTAINMENT; COST BENEFIT ANALYSIS; EVAPORATION; FISSION PRODUCTS; FUEL CYCLES; FUEL REPROCESSING; GLASS; HAZARDS; HEAT SOURCES; ION EXCHANGE CAPACITY; PUREX PROCESS; RADIOACTIVE DECAY; WASTE DISPOSAL; WASTES, LIQUID; WASTES, RADIOACTIVE; RADIATION, GAMMA

<699>

Gruber, J., Hahn-Meitner-Institut fur Kernforschung, GmbH, Berlin, German Federal Republic.

Evaluation of the Activity Levels in Fusion Reactor Blankets. Waste Problem.

KFK-B-202; Evaluation of the Activity Levels in Fusion Reactor Blankets, (Pp. 60-68), 76 pp. (1977, May)

Disposal of waste from fusion reactors is a small section in a presentation describing fusion reactor blankets. The radionuclides generated in the blanket are solid, and are fixed in the blanket unless mobilized by corrosion. Careful blanket designs are necessary to ensure that fusion reactors retain their inherent low hazard potential when compared to LWR's. One method is to shield the structural material. Other methods are the minimization of the nickel and molybdenum content, and using low activation material, such as a vanadium base alloy. Many of the design aspects are discussed in the paper. (NDV)

REACTORS, FUSION; REACTOR BLANKET; DESIGN; STAINLESS STEELS; ALLOYS; ACTIVATION PRODUCTS; NICKEL; MOLYBDENUM; VANADIUM; HAZARD ANALYSIS; HALF-LIFE, BIOLOGICAL

<700>

Jet Propulsion Laboratory, California Institute of Technology, Pasadena, CA.

Transuranics in High-Level Waste.

JPL Publication 77-59; An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management, Vol. 2, English, T.R. (Project Manager), Appendix G, (pp. G-1 - G-6), 235 pp. (1977, August 12)

This appendix compares the transuranics content of high-level wastes from three alternative LWR fuel cycles with that from the U-fueled LWR ("open-cycle"). The cases considered are: Case I, open-cycle with no reprocessing and no recycling; Case II,

<700> CONT.

U-fueled LWR with U recycling only, in which 99.5% of the U+Pu is separated from the spent fuel, leaving the fission products, 0.5% of U+Pu, and the other actinides as waste; Case III. self-generated U+Pu recycle LWR (SGR), in which both the U and Pu recovered by reprocessing spent fuel from a single LWR is recycled through a similar LWR; and Case IV. Pu recycle option, in which Pu from two LWR's is fed into a third LWR, the Pu from the third LWR being recycled back to itself. In Cases III and IV, it is assumed that 99.5% of the U+Pu is removed from the spent fuel during reprocessing. Based on calculations of mass and radioactivity content of the isotopes of Pu, Am, and Cm in the spent fuel 150 days after discharge from a typical 1000 MW(e) LWR operating under Case I, Case II shows a reduction by a factor of 200 in the quantities of Pu isotopes. However, quantities of Am and Cm are the same as for Case I. After 1000 years of decay, the curie content of the waste from Case II is reduced by a factor of approximately 30 (2900 Ci versus 91,300 Ci for Case I). Case III results in an increase in recycled Pu from an initial 204 Kg/GW(e)-yr to an "equilibrium" value of about 400 Kg/GW(e)-yr after 5 or 6 cycles. Compared to Case I, the Pu content of the spent fuel is increased by a factor of 2. However, the isotope composition is shifted toward Pu 238, 240, and 241, and away from Pu 239. The increased Pu/U ratio increases the Am and Cm in the spent fuel, making the activity of the waste higher after 1000 yrs for Case III than Case II, but still reducing activity by a factor of 8 compared to Case I. The net result of Case IV recycling is that total radioactivity of Pu in the high-level waste after 1000 years is about 10,000 Ci, and of the Am isotopes about 20,000 Ci, for a total of about 30,000 Ci/GW(e)-yr. In this case, reduction of radioactivity compared to Case I is by a factor of 3. (Auth) (LKH)

WASTES, HIGH-LEVEL; TRANSURANICS; FUEL CYCLES; REACTORS, LIGHT WATER; RECYCLING; SPENT FUELS; RADIOACTIVITY; RADIONUCLIDES; ISOTOPES; PLUTONIUM; URANIUM; AMERICIUM; CURIUM; PLUTONIUM 238; PLUTONIUM 240; PLUTONIUM 241; PLUTONIUM 239; THEORETICAL STUDIES; WASTE STORAGE; RADIATION HAZARDS

<701>
Kaser, J.D., A.K. Postma, and D.J. Bradley,
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Richland, WA.

Management of Nontritium Radioactive Wastes from Fusion Power Plants.

BNWL-2019; 53 pp. (1976, September)

Nontritium radioactive waste sources from current fusion reactor designs are discussed in this report. Disposal of large amounts of radioactive waste appears necessary for fusion reactors, but although the curie level of the waste is comparable to that of fission products in fission reactors, the isotopes are less hazardous, and have shorter half-lives. Because of this, radioactivit, from fusion power production should pose a smaller risk than radioactivity from fission reactors. Radioactive waste sources identified for the five reference plants are summarized in 4 tables. (NTIS)

WASTES, RADIOACTIVE; REACTORS, FUSION;

PREDICTIONS; WASTE TREATMENT

<702>
Keely, R.B., Allied-General Nuclear Services,
Barnwell, SC.

Description of Solid Waste Expected from a Large Commercial Reprocessing Plant.

Management of Low-Level Radioactive Waste, T.E. Carter, A.A. Moghissi, and B. Kahn (Eds.), Proceedings of a Symposium, Atlanta, GA, May 23-27, 1977. Pergamon Press, New York, NY, Ch. 2, (pp. 89-102). 1214 pp. (1979)

In this paper low-level and high-level solid wastes generated during reprocessing of nuclear fuels are characterized with respect to source, quantity, content, radioactivity, potential volume reduction, and transuranic content. Four technologies are applied, each in a separate facility, to comprise the total reprocessing complex. These technologies are: 1) separations of U, Pu, and fission products into three isolated streams, 2) conversion of uranyl nitrate to uranium hexafluoride, 3) conversion of plutonium nitrate to plutonium dioxide, and 4) solidification of the high-level liquid wastes to form a glass product. The annual volume of wastes to be expected from the reprocessing complex (in cu m) are: failed equipment - 100, solidified high-level liquids - 150, reactor scrap - 200, hulls (from chop/leach) - 650, ventilation filters - 550, process wastes - 1100, and general trash - 2250. Total waste volume = 5000 cu m/yr. Activity of solid wastes ranges from essentially zero with some general trash to around 2 X 10(E+7) Ci/cu m of solidified high/intermediate-level wastes. The lower activity process wastes are from off-gas cleanup operations in the UF6 facility, while higher level process wastes are from the UF4-UF6 conversion step and from the scrubbing of iodine from the process off-gases in the separations facility. High-level liquid wastes will accumulate at rates of about 570 L/MTU (metric ton of U) and intermediate-level liquid wastes will accumulate at about 200-400 L/MTU. These wastes will be cocalcined and vitrified to produce about 0.09 cu m/MTU of glass. Solidified high-level wastes are the only solid wastes which will release sufficient decay heat (20,000-80,000 J/cu m) to require cooling during on-site storage. Wastes are categorized as to suitability for incineration or compaction. The former include general trash, filters, process wastes, hulls, reactor wastes, high-level glass, and failed equipment; the latter include general trash, filters, hulls, reactor waste, and failed equipment. Overall volume reduction by these methods would be about 30% of the total expected waste. By careful categorization and segregation, about 80% of the reprocessing wastes should be classifiable as non-transuranic and thus disposable at a commercial burial site. The contribution of a large reprocessing complex to the volume of waste, which will have to be stored at the Federal Repository, could range from 650 to 5000 cu m/yr depending upon the reprocessors ability to sort TRU/non-TRU wastes and whether maximum volume reduction techniques are applied to the waste. (LKH)

WASTES, SOLID; WASTES, HIGH-LEVEL; WASTES, LOW-LEVEL; FUEL REPROCESSING; WASTE VOLUME; RADIOACTIVITY; SEPARATION PROCESSES;

<702>

WASTE TREATMENT

<702> CONT.
SOLIDIFICATION; VITRIFICATION; URANIUM; PLUTONIUM; FISSION PRODUCTS; URANIUM COMPOUNDS; URANIUM HEXAFLUORIDE; PLUTONIUM COMPOUNDS; WASTES, LIQUID; BOILS; FILTERS; IODINE; WASTES, INTERMEDIATE-LEVEL; CALCINATION; HEAT SOURCES; STORAGE, INTERIM; INCINERATION; COMPACTION; VOLUME REDUCTION; WASTES, TRANSURANIC; PLANTS, FUEL REPROCESSING; STORAGE, GEOLOGIC; REPOSITORY; WASTE MANAGEMENT; WASTE PROCESSING; PREDICTIONS

<703>
Logan, S.E., and G.A. Khan, University of New Mexico, Albuquerque, NM.

Selection of Significant Elements and Radionuclides for Waste Management Assessment.

Transactions of the American Nuclear Society 19:204-206; CONF-741017; American Nuclear Society Winter Meeting, Washington, DC, October 27-31, 1974, (pp. 204-206).(1974)

Calculations using the ORIGEN code indicate approximately 200 radionuclides are present at the time of separation in fuel reprocessed at 160 days. A screening method is proposed to determine the most significant elements and nuclides present in each time interval, including buildup of actinide daughters, during long-term storage. This screening method is based upon the dilution volume hazard index used extensively at ORNL. This measure, in units of cu m, is the ratio of activity of each nuclide in a given quantity of waste to the corresponding MPC value for air or water, for inhalation and ingestion hazard, respectively. Using the screening technique, at each time in the ORIGEN output tables of ingestion hazards, the 2 fission product elements, and the 3 heavy metal elements comprising the largest hazard are selected. At each time, all isotopes that collectively comprise over 99% of the hazard for each of the 3 elements from each category, which are significant at the time, are selected. The technique was applied to ORIGEN data for a reference PWR operated at an average specific power of 30 MW/MT, and burnup of 33,000 MWd/MT, for 3.3% enriched U and for self-sustaining Pu recycle fuels. Recovery of U and Pu was assumed to be 99.5%. This single screening method obtains 9 fission product isotopes of 7 elements (Sr, Cs, I, Nb, Tc, and Ru) plus Zr 93 (precursor of Nb 93m), and 18 heavy metal isotopes of 7 elements (Cm, Am, Pu, Np, Th, Ra, and Pb) for detailed study. The relatively few selected nuclides represent an average of 98.5% of the total ingestion and inhalation hazard over a range of times to 10(E+6) yr. (Auth) (LKH)

INGESTION; INHALATION; HAZARD ANALYSIS; RADIONUCLIDES; SPENT FUELS; COMPUTER CODES; WASTE STORAGE; ACTINIDES; WASTE MANAGEMENT; WASTES, HIGH-LEVEL

<704>
Phillips, J.W., and J. Grahlke, U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, DC.

Summary of Radioactivity Released in Effluents from Nuclear Power Plants from 1973 thru 1976.

EPA 52 /3-77-012; 118 pp.(1977, December)

This report is a tabular summary of radioactive wastes being released from, or shipped off-site by nuclear power plants in the U.S., and is intended as a consolidated data source for use by the EPA's Technology Assessment Division in assessing the current status of waste management, projecting future trends, identifying sources with inadequate controls, and in partially fulfilling EPA responsibilities under the Clean Air Act. This report includes plants which reached commercial operation, or had measurable releases, as of December 31, 1976, and includes only those isotopes of environmental and radiological significance. Most data were collected from plant semi-annual and annual operating reports filed with the NRC (LKH)

WASTES, RADIOACTIVE; WASTE MANAGEMENT; EFFLUENTS; EMISSIONS; NUCLEAR POWER PLANTS; PREDICTIONS

<705>
U.S. Department of Energy, Assistant Secretary for Energy Technology, Division of Spent Fuel Storage and Transfer, Washington, DC.

Spent Fuel Storage Requirements - The Need for Away-From-Reactor Storage.

DOE/ET-0075; 45 pp.(1979, February)

This report analyzes requirements for away-from-reactor (AFR) storage based upon a number of storage assumption variations concerning utility storage plans and reactor operations. This data will be used by the Department of Energy in planning storage requirements to implement the President's Spent Fuel Policy of October 18, 1977. Approximately 8000 metric tons (mT) of spent nuclear fuel is currently in storage at reactor basins around the U.S. By 1988 this number will increase to over 25000 mT, primarily from reactors already in operation. The impact on AFR requirements of variations in assumptions can be summarized in terms of five variables. Considering spent fuel pool expansion, cumulative metric tonnages of storage which would be required for 1983, 1988, and 1993 using only the existing pools are 5030, 12020, and 28950 mT, respectively. Using the latest expansion plans these values would be 560, 3860, and 18020 mT, respectively. Maximum pool expansion would reduce storage requirements to 270, 970, and 4660 mT, respectively. Assuming no transshipment, storage requirements would be 1190, 6110, and 17340 mT by 1983, 1988, and 1993, respectively. Intra-utility transshipment would reduce requirements to 560, 3860, and 18020 mT, respectively. With inter-utility transshipment, the values are 20, 190, and 10039 mT. Reactor capacities of 60%, 70%, and 80% were considered; at 60%, cumulative storage requirements are 370 mT for 1983, 2850 mT for 1988, and 9540 mT for 1993. At 70% capacity, these values are 560, 3860, and 18020 mT, respectively; at 80%, they are 800, 4600, and 19730 mT, respectively. Considering reactor discharge capabilities, with full core reserve storage requirements would be 560, 3860, and 18020 mT, respectively. Assuming annual reload only, these values are 80, 1620, and 8750 mT. Low reactor fuel burnup would require storage of 560, 3860, and 18020 mT, respectively; high burnup would reduce the values to 830, 2830, and 10800 mT. (LKH)

<705> CONT.
STORAGE, INTERIM; STORAGE, AWAY-FROM-REACTOR;
WASTE VOLUME; WASTES, RADIOACTIVE; REACTORS;
NUCLEAR POWER; WASTE STORAGE; WASTE MANAGEMENT;
TRANSPORTATION; HOLDING POND?; PREDICTIONS;
THEORETICAL STUDIES

AUTHOR INDEX

*** A ***

Aaron, W.S. 569

Abtemz, L.A. 693

Ad Hoc Committee on the Transportation of Plutonium by Air 92

Adams, P.B. 93

Ader, M. 336

Adriano, D.C. 255, 271

Aguilar, R.D. 28

Ahlstrom, S.W. 215, 222

Alberts, J.J. 255

Albrecht, E. 163, 177

Alexander, D.M. 570, 571

Alexander, C.W. 690

Allard, B. 1

Allen, E.J. 120, 896

Alpen, E.L. 216

Ames, L.L. 217, 290

Ames, L.L., Jr. 2, 191, 572

Amick, C.H. 53, 90

Anders, O.U. 3

Anderson, D.L. 389

Anderson, D.B. 304, 356

Anderson, T.J. 4

Andersson, B. 350

Andersson, L. 837

Andresen, R. 131

Andrews, L.J. 94

Angerman, C.L. 95

Angino, E.E. 351, 352, 891, 892

Antropova, Z.G. 227

Appleby, L. (Translator) 292

Apps, J.A. 218, 219

Apt, K.E. 220

Archambault, J. 5

Arnett, R.C. 215, 221, 222

Arnold, C., Jr. 96

Arnold, E.M. 289

Asaga, K. 662

Asher, J.M. (Ed.) 312, 313

Atcher, R. 289

Azesaeva, L.S. 573

*** B ***

Baca, R.G. 221

Baes, C.F., Jr. 6

Bahr, W. 574

Baker, D.A. 114, 233

Baker, W.H. 125

Balestri, R.J. 223, 894

Ballou, L. 846

Balu, K. 97

Balukova, V.D. 78, 203, 869

Bankhead, C.C. 353

Barker, J.F. 397

Barletta, R.E. 246, 336

Barlow, R.E. 461, 885

Barnes, R.M. 329

Barney, G.S. 350, 575

Barnhart, B.J. 7, 8, 9, 224

Barraclough, B. 46, 160

Barraclough, B.L. 47

Bartlett, J.V. 225, 226, 318

Batchelder, D.M. 570, 571, 622

Bates, J.K. 336

Bathke, E.A. 562

Bayhurst, B.P. 28

Bean, C.M. 336

Beeten, W.A. 576

Belkin, H.E. 72

Bell, M.J. 691

Belova, V.I. 227

Belter, W.G. 355

Bensen, D.W. 286

Berbano, M.C. 279

Bernstein, G.J. 336

Berreth, J.R. 602

Bertozzi, G. 256

Bevier, M.L. 810

Beyer, C.E. 110

Bibler, N.E. 98, 99, 100, 577

Biggers, J.V. 10

Bind, G.M. 10

Binnall, F.P. 538

Bischof, W. 895

Bishop, W.P. 356, 588

Blakeman, E.D. 120, 896

Blalock, S.D., Jr. (Translator) 229

Blanco, R.E. 238, 500, 501, 502, 521, 596, 652, 695

Blane, D.M. 570

Blomke, J.O. 579, 690

Boch, R.L. 529

Bochkarev, V.M. 227

Bougly, W.J., Jr. 359, 360, 530, 531

Bogorov, V.G. 357

Boissonnet, B. 281

Bond, W.D. 579

Bond, W.H. 634

Bondietti, E.A. 11, 228

Bonner, W.F. 580

Bopp, C.D. 82

Borchardt, J. 635

Borg, I.Y. 163

Boyard, P. 229

Bowen, V.T. 230

Boyd, R.D. 532

Bradley, D. 315

Bradley, E.J. 101, 153, 358, 369, 661, 701

Bradley, B.P. 683

Bradshaw, R.L. 12, 13, 359, 360, 361, 531, 533

Bradshaw, R.L. (Ed.) 316

Braithwaite, J.W. 102, 103, 148

Brannan, J.P. 898

Braun, C. 601

Braun, R.L. 898, 632

Brebano, M.C. 259

Bredehoeft, J.D. 362

Breest, H.C. 897

Brewer, L.W. 363

Brezneva, N.E. 108

Briggs, L.I., Jr. 364

Brindley, G.W. 14

Brinkley, F.S. 675

Brookins, D.C. 15

Brookins, D.G. 229, 231

Brown, D.J. 198, 221, 268

Brown, D.L. 365

Brown, P.M. 365, 366

Brown, P.J. (Ed.) 367

Browning, R.F. 665
Brunton, G.D. 368
Bryant, E.A. 220
Budnitz, R.J. 512
Bunnell, L.R. 661
Burenkov, E.P. 581
Burkholder, H.C. 232, 233, 234, 369, 582
Burkholder, H.C. (Ed.) 330
Burleigh, R.R. 534
Burris, L. 336, 672
Byerly, D.W. 370, 371

*** C ***

Caldwell, D.E. 7, 8, 9
Campbell, A.V. 584
Campbell, E.W. 7, 8, 9
Campbell, J.E. 506
Campbell, N.H. (Ed.) 582
Campbell, R.J. 301
Cannon, T.F. 336
Carlsson, H. 535
Carruthers, J. 318, 372
Carter, W.L. 373
Cash, D.L. 136, 667
Cassano, G. 403
Cerny, P. 105
Chambers, W.F. 80
Chen, T. 374
Chapin, J.A. 164, 583, 692, 693
Chapman, W. 375
Charles, R.W. 149
Chen, H. 126
Cheng, L. 235
Cherry, J.A. 376
Childs, E.I. 647
Choi, E.C. 584
Christenson, C.W. 678
Chung, S.F. 490
Claassen, H.C. 88, 377
Claes, J. 682
Claiborne, H.C. 319, 585
Clasen, J. 676
Cleniger, H.O. 233
Close, D.A. 134
Clymer, H.A. 16
Cohen, D. 250
Cohen, J.J. 84, 498, 632
Cohen, J.L. 17
Cohen, P. 165
Cole, C.R. 215, 222
Cole, H.A. 106
Cole, H.S. 602, 629
Coles, D.G. 153
Collins, J.T. 691
Colombo, P. 166, 167, 168
Colton, G.H. 379
Cooper, W. 202
Cook, H.G.H. 18, 19, 210, 378, 380, 381, 382, 614
Cooper, K.C. 107
Cooper, N. 150
Corey, J.C. 271
Couture, H.A. 297, 336
Covan, C.E. 289
Covan, G.A. 236
Crandall, J.L. 596
Crawley, D.T. 170
Croff, A.G. 690
Crouch, S.L. 20
Crove, H.H. 77

*** D ***

D'Alessandro, H. 256
d'Entremont, P.D. 478
Dahlman, R.C. 652
Daimon, H. 662
Dalton, J.T. 608
Daniels, W.R. 28, 244
Dau, G. 587
Davidson, R.H. 320
Davis, W., Jr. 238, 500, 501, 502, 690, 695
Dawson, P.R. 25, 386
de Josselin de Jong, G. 239
De, A.K. 635
Defigh-Price, C. 234
Degerman, O. 687
Deju, R.A. 222, 387, 388, 389, 390
DeLagunn, W. 391
Dennis, A.G. 537
DeSteese, J.G. 543
Devarey, E.E. 62
Diefenbacher, K. 177, 202
Dillich, S. 133
Dillon, R.L. 598
Dimuzio, J.J. 539
Doctor, P.G. 26
Dolgov, V.V. 392
Dosch, R.G. 280, 281, 636
Doty, J.W. 570, 571
Dreschhoff, G. 352
Drobnik, S. 574
Drolet, T.S. 584
Droste, J.B. 393
Dube, A. 36
DuBois, A.O. 534
Duckworth, J.P. 589
Duffy, C.J. 29
Duguid, J.O. 242, 243, 503
Dumont, G. 676, 682
Dupree, S.A. 590
Duran, O. 837
Dyroff, H. 696

*** E ***

Eccleston, G. 134
Eckerman, K.F. 253
Edmund, R.W. 398
Eliason, J.R. 27
Elliot, H.H. 108
Empson, F.W. 109, 328, 359, 360, 395, 539
Engelmann, W.J. 177
England, A.W. 362
English, T.D. (Project Manager) 321
Enzminger, D.A. 110
Erdal, B.R. 28, 280
Erdal, B.R. (Ed.), et al. 171
Ewest, E. 591
Ewing, B.R. 29
Ewing, K.C. 30, 111

*** F ***

Fahundiny, B.B. 173
Farmer, B.C. 472
Farrell, W.E. 481
Fecht, E.D. 193, 194, 195, 158
Feese, H.L. 592
Feher, L. 593
Feng, P.K. 349
Fernandez, S.J. 172, 205, 590
Ferrell, R.e., Jr. 485
Fey, P.L. 595
Fickies, B.E. 173
Filer, T.D. 181
Finney, B.C. 238, 500, 501, 502, 529, 652, 695
Finsterwalder, W. 578
Fitzgerald, C.I. 596
Fleischmann, F.K. 696
Flynn, E.F. 286, 336
Fonin, Yu.E. 626
Fong, L.Q. 490, 616
Foote, H.P. 215
Forberg, S. 597
Forbes, I.A. 582
Fossum, A.F. 589
Francis, A.J. 166, 167, 169
Francis, C.W. 11, 228
Frank, F.J. 189
Frejacques, C. 287
Frendberg, R.I. 277
Fried, S. 31, 32, 288, 289, 250, 251, 299
Fried, S.H. 252, 299
Friedman, A. 248, 251
Friedman, A.W. 31, 32, 289, 250, 252, 298, 299
Friedman, E.M. (Ed.) 330
Frigerio, N.A. 253
Fritz, P. 397
Fulkerson, S.D. 617
Fullwood, R.R. 565
Fussell, J.B. 582

*** G ***

Gaillodreau, C. 165
Gale, J.E. 90, 329, 397, 398, 470
Galley, J.E. 399
Galley, J.E. (Ed.) 400
Gangwer, T. 601
Garbarini, G.S. 481
Garland, T.B. 370
Gaudette, H.E. 33
Geble, R.R. 472
Gemmell, L. 598
Gephart, R.E. 222
Gere, F. 38, 254, 402, 403, 404
Gerads, B. 139
Gerding, T.J. 336, 672
Gibbons, J.W. 255
Gillham, R.W. 376
Girardi, F. 256
Girton, R.C. 594
Giuffre, M.S. 257
Gloyna, E.F. 311
Gnirk, P.F. 174
Godbee, H.W. 258, 586, 599, 621
Goebel, E.D. 398
Golchert, N.W. 600
Goldber, S.M. 259
Goldstein, M. 601
Gombert, D. 602
Goodlett, C.B. 603
Goodman, P.R. 373
Gore, B.F. 697
Grass, W.H. 581
Granquist, D.P. 698
Grasty, A. 229
Gray, D. 375
Gray, W.J. 604, 661
Greenberg, J. 369, 582
Greene, R.B. 59
Gregory, W.T., III 552
Griggs, R. 588
Grim, R.E. 33
Grimwood, P. 260
Grimwood, P.D. 261, 268
Grisak, G.E. 376
Grover, J.R. 108, 405
Gruber, J. 609
Gruhlke, J. 700
Grutzeck, M.W. 156, 406
Guber, H. 605

*** H ***

Haden, W.L., Jr. 606
Haggblom, H. 252
Hagstrom, E. 424
Hakonson, T.E. 293
Halaszovich, S. 607
Hale, W.H. 683
Halen, P.A. 350
Hall, A.P. 608
Hall, R.J. 112
Halleck, E.H. 37
Hallett, R. 7, 9
Hallett, R.E. 8
Hamilton, M. 288
Hanner, R.L. 682
Hanstra, J. 263, 407
Hanson, D.W. 86
Hardaway, J.E. 409
Hardin, J.M. 7, 9
Hardt, T.L. 331
Hardwick, W.H. 139
Harmon K.M. 175
Hartwig, C.M. 113
Hatch, L.F. 609
Haug, H.O. 35, 409
Haydon, M.P. 6.2
Heaberlin, S.W. 118
Headley, T.J. 115, 122
Healey, J.T. 115
Heard, W.C. 36
Heath, C.A. 530
Hebel, L.C. 610
Heckman, R.A. 535
Heckman, R.C. 569
Heiken, G.H. 610
Heimerl, R. 611
Heinonen, J.S. 653
Heiskell, W.H. 187
Hendricks, J.A. 158
Herald, W.R. 578, 621
Hermans, P. 251

Hermansky, B. 322
Hetzer, D.C. 285
Hickey, J.B.B. 633
Hickox, C.E., Jr. 37
Hild, W. 574, 612
Hill, G.S. 238, 500, 501, 502, 521, 695
Hill, J. 613
Hill, L.R. 328
Hill, R. 260
Hill, R.D. 260
Hilpert, K. 139
Hine, B.E. 168, 582, 692, 693
Hines, J. 248, 249, 251
Hines, J.J. 252
Hinkebein, T.E. 38, 39
Hirling, J. 593
Hite, B.J. 811
Hlava, P.F. 38, 39, 40, 115, 122
Hoda, B. 861
Hodge, C.V. 506
Hodge, V.F. 235
Hoffman, D.C. 244
Hoffman, T.L. 116, 117
Holcomb, B.D. 566
Holdsworth, T. 505
Hollister, C.D. 356
Holmberg, K.E. 276
Holoway, C.F. 323
Homstead, J.F. 265
Hood, H. 381
Horn, S.J. 158
Hoskins, E.B. 178
Houser, B.I. 109, 323, 675
Houser, C.A. 178
Huckaba, W.A. 812
Hudson, B. 608
Hudson, C.B. 507
Huge-Fersson, T. 822
Hull, H.L. 683
Hussain, M. 635
Hustrulid, W. 837

*** I ***

Inoue, Y. 266, 288
Isaacson, B.E. 268
Isherwood, D. 269
Ito, A. 639
Iwai, B. 90

*** J ***

Jacobs, D.G. 675
Jain, B.K. 97
Jakubick, A.T. 270
Jamison, D.B. 474
Jamison, D.K. (Ed.) 696
Janzen, G. 119, 233, 293, 294
Jardine, L.J. 286, 336, 618
Jenkins, I.L. 416
Jenkins, J.D. 120, 496
Jenks, G.H. 81, 82, 121
Jennings, A.S. 615
Johnson, A.J. 490, 616
Johnson, J.B. 617
Johnson, K.D.B. 405
Johnson, K.S. 817
Johnson, L.F. 618
Johnson, L.J. 652
Johnson, W.A. 505
Johnstone, J.K. 122, 636
Jones, P.H. 818
Joy, D.S. 147, 258, 566

*** K ***

Kabakchi, S.A. 78
Kabele, T.J. 697
Kahl, L. 574, 605
Kanno, T. 619
Kaplanek, D.W. 865
Karlsson, D.W. (Editor) 176
Kaser, J.D. 119, 697, 701
Kasza, J. 593
Katayama, Y.B. 661
Kaufman, R.J. 266
Kazanjian, A.R. 83
Kee, C.W. 690
Keely, R.B. 702
Keenan, T.K. 819
Kelley, J.A. 84, 99, 868, 620,
Kelley, J.A. 659, 683
Kelm, H. 574
Kemper, P.S. 598
Kenna, B.T. 128, 636
Kerrigan, J.V. 29
Khonikevich, R.A. 581, 628
Kibbe, R.R. 529
Kibbey, A.H. 258, 621
King, F.D. 125
Kipatsi, H. 9
Kirkham, R.R. 271
Kittrick, J.A. 301
Kiyose, R. 509
Klaffky, R.W. 91
Klechkovsky, V.N. 85
Klechkovsky, V.N. (Ed.) 272
Klingler, L.R. 622
Klockars, C.E. 276
Klager, W. 574, 612
Knapp, G.S. 126
Knapp, R.R. 273, 274, 275
Knecht, D.A. 181
Knoll, K.-H. 286
Knorovsky, G.A. 183
Kobisk, E.H. 569
Kochen, R.L. 688
Koenst, J.W. 623
Kohorin, I.N. 203
Kolb, C.R. 820, 828
Kolditz, H. 177
Kolychev, B.S. 392, 628
Komarneni, S. 58, 282
Kondrat'ev, A.N. 573
Konstantinovich, V.V. 392
Konzek, G.J. 326
Koplik, C.M. 583
Korner, W. 450
Korthaus, E. 202
Kosiewicz, S. 46, 160
Kosiewicz, S.T. 87
Koster, R. 668
Koots, H.J.C. 512
Krause, H. 177, 574, 612, 625
Kreps, E.H. 357
Krumhansl, J.L. 821

Krumpelt, R. 336
Kryukov, I.I. 573
Kubota, H. 359, 366
Kuhn, E. 177
Kul'bashnaya, E.V. 626
Kulichenko, V.V. 392, 573
Kullen, B.J. 336, 672
Kumar, S.B. 485
Kumra, M.S. 523
Kupfer, R.J. 115, 627, 669
Kurfurst, P.J. 422
Kunzi, J. 653
Kurbatsov, Yu.V. 689

*** L ***

La Porte, Z.M. 510
Lacy, W.J. 178
Lau, D.J. 126
Laub, E.M. 628
Laubert, S.J. 338
Landstrom, O. 276
Lappin, R. 423
LaSiviere, D.C. 127
Larker, H. 597
Larker, H.T. 629
Larson, D.W. 50
Lazer, M. 607
Lash, T.E. 511
Laude, P. 128
Laurence, F.G. 28, 288
Laurence, W.C. 630
Lazarev, L.N. 689
Leachman, R.E. 588
Leddicotte, G.W. 277
Lee, J.I. 631
Lee, P.C.Y. 282
Lemoine, J. 5
Lerrmann, W.I. 327
Lenzi, G. 803
Leonard, R.A. 336
Leontichuk 203
Lester, R. 601
Levine, M.S. 128
Levitz, N.M. 672
Levy, E.B. 163, 278
Levy, P.W. 81, 82
Lewin, R.A. 235
Lewis, A.B. 632
Lewis, A.J. 420
Lewis, E.L. 570, 571, 622
Lewis, M.W. 512
Liljemin, J.O. 420
Lin, W. 48, 49
Lincoln, B.C. 50
Lindauer, R.E. 652
Linderoth, C.E. 179, 193
Lini, D.C. 129
Lloyd, O.N., Jr. 365
Loewenstein, W.B. 512
Logan, S.E. 259, 279, 703
Logsdon, J.E. 280, 633
Loleit, A.J. 443
Lomenick, T.F. 51, 328, 425, 426, 427
Longo, J.M. 637
Lucas, J. 279
Lush, D.L. 376
Lusso, F. 562
Luthy, D.F. 634
Lutze, W. 635
Lutze, W. (Ed.) 330
Lynch, R.W. 231
Lynch, R.W. 636
Lyubtsev, B.I. 689

*** M ***

Maas, E.T., Jr. 637
MacMillier, N.H. 130
Madewell, D.G. 323
Maestas, S. 29
Magnani, N.J. 102
Magnusson, K.A. 437
Mahara, Y. 248
Malaro, J.C. 513
Malov, V. 131
Manin, A.V. 567
Mandel, S. 119
Manning, T.A., Jr. 472
Marine, I.W. 180
Markham, C.D. 181
Marlow, J.A.C. 608
Marshall, J.S. 201
Martelle, L.L. 647
Martin, J.E. 514, 515
Martin, J.J. 281
Martin, R.T. 52, 66
Martinez-Baez, L.F. 53
Martinez, E. 7, 8, 9
Martinez, J.D. 420, 461
Martynov, Y.P. 392, 573
Maslovskii, E.E. 182
Maslyanovskii, R.V. 567
Mason, N.P. 651
Mason, R.J. 300
Mather, J. 375
Mather, J.R. 306
Mathers, R.G. 638
Mather, A.K. 219
Matsuzuru, H. 639
Mattsson, E. 132
Matveev, L.N. 624
Mayer, D.W. 289
Mayman, S.E. 329
McCann, T.P. 429
McCarthy, G.J. 54, 282, 580, 665
McCarthy, G.J. (Ed.) 330
McClain, W.C. 20, 55, 361, 368, 395, 528, 539
McClain, W.C. (Ed.) 316
McDaniel, E.W. 59, 218, 683
McDonell, W.R. 133
McElroy, J.L. 648
McEvoy, H.B. 56
McHenry, J.R. 193
McKinley, K.B. 184, 185, 186
McKinney, J.D. 184, 185, 186
McKinstry, H.A. 57
McLeod, K.W. 255
McMillan, T.S. 868
McWay, T.W. 617, 642
Meehan, W.F. 331, 672
Medvedeva, N.L. 78
Meile, L.J. 647
Meisenhelder, J.H. 336

Mellen, P.F. 330
Mellinger, G.B. 661
Mendel, J.E. 643, 648, 661
Menlove, H.O. 138
Menon, S.N. 135
Mensing, R.W. 153
Menzel, R.G. 58
Mercer, J.S. 437
Merlini, R.J. 136
Merrill, R.H. 373
Merritt, W.F. 605
Merz, E. 607
Metcalf, J.H. 363
Metzger, C.F. 33
Metzger, J.C. 187
Meyer, F.G. 349
Mieza, F.R. 281
Miller, B.J. 420
Milloy, J.B. 537
Mimura, H. 619
Minaev, A.A. 104
Miyake, Y. 188, 302
Molecke, M.A. 103, 137, 300
Moore, D.A. 70
Moore, J.G. 59, 60, 62, 218, 258
Moore, J.G. (Ed.) 330
Moore, R.E. 218, 500, 501, 502, 521, 695
Morgan, J.W., Jr. (Ed.) 646
Morgan, R.T. 59, 61, 62, 63
Morisawa, S. 284
Morris, J.B. 405
Morton, R.D. 277
Morton, R.J. 331
Moss, W. 568
Motes, B.T. 172, 205
Moehlberger, W.R. 432
Mullen, W.P. 560
Murphy, K.D. 124
Murphy, L. (Translator) 466
Myles, K.M. 336

*** N ***

Naeser, C.W. 64
Nakamura, J.W. 306
Nakamura, R. 285
Navratil, J.D. 647, 648
Nelson, D.C. 127
Nelson, J.L. 151, 286, 649
Nelson, P. 437
Nelson, P.G. 580
Neretnieks, I. 287
Newcherch, E.J. 428
Newcomb, R.C. 189
Newman, E. 60, 62
Nielsen, S.O. 517
Nieschmidt, E.B. 650
Nikiforov, A.S. 392
Nikipelov, B.V. 392
Nishita, H. 288
Norgren, D.W. 524
Norris, A.E. 65
Nowak, E.J. 636
Nugent, V. (Translator) 5

*** O ***

Oden, S.L. 472
Odoj, R. 138
Oertel, G.K. 190
Oliver, P.Q. 28
Olivier, J.P. 404, 518
Olsen, J.R. 66
Ondracek, G. 605
Ondrejcin, R.S. 139
Onishi, Y. 289
Orebaugh, E.G. 100
Orr, B.R. 431
Ortiz, A.P. 534
Oston, S.G. 110
Oziramer, S.W. 104, 302

*** P ***

Pahva, S.B. 494
Paille, L.K. 472
Palmer, S.P. 481
Panter, H.R. 651
Parisoeau, W.G. 67, 180, 178, 480
Parker, F.L. 359, 360
Parker, H.E. 327
Paterson, E.S.B. 481
Paulsson, B. 437
Pearce, D.W. 179, 191
Pechin, W.H. 652
Peckham, P.E. 192
Peeters, E. 476, 692
Peister, E.B. 529
Pelto, P.J. 542
Pelto, R.H. 336
Peltonen, E.K. 653
Pelzer, W. 495
Pentz, D.L. 583
Perona, J.J. 359, 360, 685
Perritt, R.W. 472
Perry, E.R. 520
Perzl, F. 161
Peterson, J.A. 483
Peterson, P.L. 114
Petrie, G.M. 560
Pfaffelhuber, J. 497
Pfoertsch, D.E. 640
Phillips, J.W. 704
Pierce, G.D. 172, 205
Pierce, R.E. 654
Pilger, R.H., Jr. 485
Pimenov, N.K. 203
Pinchback, T.R. 141
Piper, A.M. 68, 444
Pittman, F.K. 551
Platt, A.E. 333, 519, 644
Plodinec, M.J. 655, 656, 657, 658
Podio, A.L. 273, 274, 275
Pomiak, G.S. 154
Pornink, G.S. 204
Port, E.A. 142
Postma, A.K. 701
Potter, J.W. 69
Potter, R.W., II 16
Potter, R.W., II, (Ed.) 330
Potter, R.W., Jr. 80
Powers, D.W. 334
Prenger, F.C. 107
Price, G.H. 290
Price, V.B. 19[illegible], 194, 195, 198
Privrasky, N.C. 429

AUTHOR INDEX

Pophal, K.W. 381

Portyoun, W.D. 196

Pot, H. 291

*** Q ***

Quarterman, L.A. 252

Quimby, F.C. 569

*** R ***

Rachiele, R. 437

Rack, H.J. 183

Rai, D. 70, 71, 217, 300

Rakov, D.A. 202

Ramachandran, S. 97

Ramspott, L. 486

Ramspott, L.D. 163

Ramspott, L.D. (Ed.) 487

Rancon, D. 292

Rankin, W.D. 95, 620, 659

Ranschnig, D. 495

Ravindranath, V. 523

Raymond, J.R. 369

Reeves, H. 203

Regan, W.W., Jr. 689

Reichert, R.G. 488

Reid, H.S. 365, 366

Relyea, J.F. 71

Revelle, R. 449

Rex, R.W. 678

Rhoads, R.E. 564

Rhodes, D.W. 188, 631, 687

Richter, D. 450

Rickert, P. 288

Rickert, P.G. 298, 299

Rindfleische, J.R. 660

Ringwood, A.E. 185

Roberts, F.P. 661

Robinson, R.V. 293, 298

Rodger, W.A. 277

Roedder, E. 72

Rogers, G.C. 60, 62

Rogers, M.B. 196, 452

Ross, B. 257

Ross, W.A. 580, 661

Rotty, R.M. 520

RCutson, R.C. 197, 198, 293, 294

Rovik, J.E. 461

Rowe, W.D. 512

Roy, R.R. 156, 282, 662

Roy, R. 130, 282, 663

Rudolph, G. 422, 664

Rusin, J.M. 665

Russell, J.E. 174

Ruth, W. 574

Rydberg, J. 1

*** S ***

Sahau, C. 251

Saidl, J. 605

Sample, C.R. 326

Sanford, R.F. 329, 470

Sarubashi, K. 188, 303

Saunders, R.B. 336

Saunders, D.F. 491, 492

Schaefer, R.B. 489

Schaffer, W.F. 109, 530, 531

Scheetz, B.E. 58, 156, 282, 662

Schlegal, R.L. 493

Schmelz, D.L. 200

Schneider, R. 503

Schofield, J.S. 158

Schroef, T. 437

Schreiner, F. 288

Schrieber, D.L. 302

Schuchardt, H.C. 177

Schulz, W.W. 627, 669

Schwartz, L.L. 498

Schweiger, J.S. 153

Schwoebel, R.L. (Ed.) 330

Scully, L.W. 537

Seeland, O.H. 675

Sears, H.W. 521

Sedlet, J. 600

Sedov, V.M. 392

Seefeldt, W.B. 336, 672

Seitz, M.G. 297, 298, 299, 336

Sensi, L. 403

Serne, R.J. 70, 71, 215, 289, 300, 369

Shaffer, S.E. 330

Shapiro, S.R. 670

Shappert, L.B. 107

Shashukov, E.A. 689

Shatsillo, V.G. 392

Shaw, H.R. 562

Shefelbine, R.C. 537

Sheppard, J.C. 301

Shoup, R.L. 522

Sicrek, A.A. 336

Singh, N. 201

Sinha Ray, R.K. 523

Sisson, C.E. 50

Sjoblom, R. 288

Slansky, C.R. 671

Smailes, E. 177, 202

Smith, R.J. 659

Smith, C.G., Jr. 620, 669, 661

Smith, C.H. 687

Smith, D.T. 76

Smith, D.K. 292

Smyrl, W.H. 189

Smyth, J.F. 77, 189

Snow, R.H. 462

Snyder, R.M. 187

Sokol, J. 150

Sokolova, L.N. 85

Soosseliar, Y. 663

Speir, L.G. 130

Spencer, C.W. 683

Spitsyn, V.I. 78, 203, 668

Spriet, H. 476, 692

Staley, G.H. 302

Stanley, R.G. 665

Staples, E.A. 208

Starfield, A.M. 55

Stead, F.I. 829

Steen, A.J. 288

Steinhorn, T.L. 17

Steindler, M.J. 286, 298, 299, 336, 618, 672

Stephansson, O. 666

Stephenson, L.L. 188

Stevenson, J.D. (Ed.) 686

Stewart, D.B. 79, 80, 362
Stewart, E.T. 508
Stinebough, D.E. 537
Stohl, F.V. 122
Stone, J.A. 467, 469, 673, 674, 693
Stone, R. 163
Story, G.L. 670
Stottlemyre, J.A. 369, 542, 550
Stone, P.H 253
Strachan, D.H. 115, 627
Strand, J.P. 189
Straub, C.P. 337, 338, 339
Strickert, B. 248
Strickert, R.G. 250
Stromatt, R.R. 70
Strumness, E.G. 359, 360
Sugiura, Y. 303
Sullivan, D.F. 493
Sumner, W.B. 205
Sunder Rajan, N.S. 523
Sundquist, J.E. 151
Sutherland, A.A., Jr. 110
Suzuki, Y. 285
Svantesson, I. 420
Swan, G. 469
Swyler, K.J. 81, 82

*** T ***

Talbert, D.M. 304
Talbot, R. 583
Tamamaki, H.Y. 470
Tamura, T. 305, 675
Tarnuzzer, E.C. 277
Tatada, K. 136
Teter, A.R. 455
Teves, M.A. 498
Thiele, D. 607
Thielemann, H 177
Thomas, H.C. 676, 677
Thomas, K.T. 523
Thomas, L.I. 205
Thomas, P.G. 678
Thompson, F.I. 289
Thompson, G.R. 607, 608, 683
Thompson, J. L. 83
Thoms, R.L. 420, 461, 472
Thornthwaite, C.E. 306
Tierney, R.S. 562
Tillerson, J.P. 25, 386
Tolman, C.R. (Ed.) 693
Tomlinson, H. 473
Tomlinson, P.E. 690
Tonnesson, K.A. 84
Topp, S.V. 474
Torres, D.V. 223, 494
Torstenfelt, B. 1
Trask, R.J. 362
Trevorrow, L.E. 336, 672
Tsang, C.F. 374
Tsao, L. 219
Tselishcheva, G.N. 45, 272
Tsong, I.S.T. 119
Tuney, H.T. 472
Turcotte, R.P. 155, 661, 674
Turi, G.P. 302

*** U ***

Ueda, T. 285
Uffen, R.J. 527
Ullrich, P.A. 483
Unione, A.J. 565

*** V ***

Van de Voorde, H. 476, 682
Van Koynenburg, R.A. 110
Veal, B.W. 126
Veal, M.K. 401
Vegorn, S.H., Jr. 650
Vidale, R.J. 189
Vilks, G. 477
Vogler, S. 336
Voight, B. 85
Von Hippel, F. 512
Voss, J.P. 582

*** W ***

Wade, E.L. 204
Wald, J.W. 661, 679
Wallace, R.H. 48, 683
Wallsbeg, H.A. 212
Walsh, E.L. 563
Walters, R.P. 478, 479
Walton, R.D., Jr. 604
Warde, J.P. 617
Warner, D.L. 480
Waters, K.H. 481
Waversik, W.R. 86, 174
Weart, R.P. 334, 482
Weaver, C.E. 87, 483
Webb, G. 260
Webb, G.A.M. 261
Weber, K.E. 661
Webster, L.S. 336, 672
Weeber, P. 31
Weed, H.C. 153
Weerr, H.O. 213, 214, 601, 605
Weisenberger, S. 686
Weiss, A.J. 166, 167, 168
Wenner, D.B. 494
West, G.A. 59, 63
West, P.J. 327
Westerberg, S. 276
Westermark, T. 597
Westra, D.G. 150
Westsik, J.H., Jr. 155, 661
Whan, G.A. 703
Wheeler, R.L. 196
Whisnan, J.K. 472, 485
White, A.F. 88
White, R.L. 156
White, J.G. 490
White, R.P. 54, 85, 118, 282, 600
Widell, B. 597
Wilcox, R.E. 428, 485
Wilding, R.W. 687
Wildung, R.E. 310
Wiley, J.B. 186, 608
Williams, [illegible] 72
Williams, R. 587
Wilson, J.E. 829
Winchester, J.W. 157
Winegardner, D.K. 582

Winograd, I.J. 362

Winsch, I.O. 672

Witherspoon, J.P. 236, 500, 501, 502, 521, 652, 695

Witherspoon, P.A. 19, 90, 210, 282, 390, 487

Wodrich, D.B. 158

Wolfe-Confer, B. 662

Wolfsberg, K. 28, 83, 284

Wovak, W.E. 537

Wrangles, G. 159

*** Y ***

Yardley, D.H. 489

Yousef, Y.A. 311

Yudin, P.P. 283

*** Z ***

Zachariasen, F. 512

Zeigler, D.L. 490

Zeller, E.J. 352, 491, 492

Zeslyanukhin, V.I. 609

Zerty, C.D. 528

Zerweck, A. 46, 47, 91, 160

Ziegler, A.A. 336

Zimerman, K.G. 156

Zweifel, H.A. 127

KEYWORD INDEX

*** A ***

A ZONE 305

ABSORPTION 22, 85, 197, 200, 208, 251, 272, 283, 299, 350, 502, 570, 593, 601, 613, 681

ACCIDENTS 92, 95, 114, 123, 147, 152, 162, 168, 190, 221, 255, 277, 278, 279, 296, 307, 385, 336, 343, 346, 347, 433, 445, 478, 505, 506, 512, 527, 530, 552, 558, 564, 565, 568

ACCIDENTS, OCCUPATIONAL 382

ACCRETION 356

ACIDS 58, 133, 178, 349, 416, 424, 583, 687, 689, 672, 678, 681

ACIDS, ORGANIC 96, 167, 495, 625, 687

ACTINIDES 1, 6, 15, 32, 35, 58, 131, 151, 219, 231, 249, 256, 252, 256, 259, 290, 300, 301, 315, 317, 336, 346, 347, 351, 358, 369, 375, 402, 409, 416, 424, 433, 434, 522, 574, 575, 585, 588, 590, 601, 625, 646, 643, 664, 667, 668, 679, 690, 703

ACTINIUM 261

ACTINIUM 227 43

ACTIVATED CARBON 320

ACTIVATION ENERGY 679

ACTIVATION PRODUCTS 259, 280, 699

ADSORPTION 31, 32, 70, 157, 165, 178, 193, 191, 217, 218, 228, 241, 249, 252, 280, 300, 301, 337, 360, 368, 375, 471, 570, 576, 606, 655

ADULTS 230, 500

AEROSOLS 187, 270, 506

AGE DETERMINATION 74, 330, 376, 397

AGRICULTURE 222, 255, 277

AGRILLITES 29

AIR 125, 230, 329, 346, 347

AIR QUALITY 308

AIRITE 606

ALGAE 235, 285

ALKALINITY 110, 168, 180, 213, 397, 609

ALLOYS 95, 102, 103, 116, 117, 143, 148, 159, 200, 569, 623, 699

ALLUVIUM 83, 189, 197, 831

ALPHA PARTICLES 67, 83, 91, 98, 99, 100, 120, 133, 166, 186, 187, 197, 256, 271, 307, 309, 327, 330, 339, 355, 377, 413, 419, 509, 526, 581, 577, 599, 613, 643, 659, 672, 678, 696

ALUMINUM 112, 116, 117, 118, 150, 200, 208, 460, 606, 656, 657

ALUMINUM COMPOUNDS 132, 620, 629,

ALUMINUM COMPOUNDS 658

AMERICIUM 15, 157, 178, 181, 208, 249, 250, 252, 298, 299, 301, 351, 405, 424, 595, 598, 599, 663, 668, 700

AMERICIUM COMPOUNDS 159, 255

AMERICIUM 241 67, 83, 369, 208, 230, 256, 261, 264, 290, 293, 307, 309, 310

AMERICIUM 242 [illegible]

AMERICIUM 243 256, 269, 310

AMMONIA 187

AMALCITE 686

AMBERITE 16, 50, 78, 384, 411, 413, 425, 432, 461, 478

ANIMALS 197, 271, 307, 308, 310, 389

ANIMALS, BURROWING 268

ANIONS 178, 197, 217, 269

ANISOTROPY 21, 69, 291, 469, 535

ANORTHOSITES 372

ANTIMONY 125 620

APATITE 189

AQUIFERS 73, 189, 192, 216, 222, 241, 242, 263, 267, 268, 275, 300, 302, 309, 353, 377, 387, 388, 417, 418, 830, 831, 856, 859, 861, 870, 879, 880, 894, 505, 536

AQUIFERS, CONFINED 197, 203, 267, 291, 348, 366, 408, 860

ARGILLITES 83, 396

ARGON 398

ARSENIC 84

ASHES 159, 309, 570, 577, 588, 622, 625, 638, 687

ASHPLOWS 196

ASPHALT 208, 338, 593, 596, 621, 685

ATMOSPHERE 197, 567

ATTAPULGITE 213, 606, 681

AUTORADIOGRAPHY 83, 216, 229, 288

*** B ***

BACKFILLING 383

BACKGROUND RADIATION 138, 253, 308, 309, 346, 387, 363

BACTERIA 7, 8, 9, 220, 310

BARIUM 58, 88

BARIUM 133 28

BARIUM 137M 590

BARIUM 140 144

BASALTS 11, 23, 24, 31, 32, 53, 54, 63, 74, 75, 77, 189, 192, 193, 194, 195, 200, 210, 219, 220, 227, 249, 250, 252, 282, 299, 313, 335, 342, 355, 356, 373, 383, 385, 387, 388, 389, 390, 406, 413, 419, 425, 482, 444, 451, 458, 546, 549, 552, 553, 558

BASES 178

BASINS, ARTESIAN 387, 505

BASINS, DRAINAGE 189, 420, 428

BASINS, STRUCTURAL 240, 364, 379, 385, 393, 394, 399, 400, 401, 458, 412, 417, 429, 432, 438, 443, 452, 456, 480, 551

BEDROCK 180, 190, 217, 351, 355, 366, 371, 393, 402, 405, 471, 478

BEETROS 261

BENTONITES 132, 150, 276, 302, 575, 628, 660, 670

BERMS 285

BETA PARTICLES 120, 164, 166, 179, 191, 193, 205, 280, 330, 370, 363, 377, 670, 698

BIBLIOGRAPHIES 163, 323, 326, 367, 436, 465

BINDERS 196, 569, 575, 620

BIOLOGICAL TRANSPORT 235, 255, 285

BIOSPHERE 201, 234, 250, 278, 335, 367, 405, 560, 633

BIOTA 181, 201, 255, 261, 311

BITUMEN 59, 59, 159, 202, 256, 369, 392, 501, 612, 625, 628

BITUMINIZATION 476, 523, 574, 592, 612, 628, 626, 653, 682

BODY BURDEN 230, 271, 283, 339, 500

BONE CHAR 571, 578

BONES 230, 271, 309, 500, 501

BOREHOLES 35, 52, 59, 66, 132, 140, 196, 267, 273, 274, 276, 312, 313, 319, 340, 365, 381, 387, 388, 389, 390, 397, 398, 410, 411, 413, 422, 425, 425, 426, 437, 459, 479, 535, 536, 628

BORIC ACID 528

BORON 128, 130, 159

BORON COMPOUNDS 131, 602, 604, 607, 608

BORES 136, 180

BREAKTHROUGH DISTRIBUTION 29, 60, 178, 191, 197, 200, 272, 276, 286, 573, 638, 689

BRINES 15, 16, 41, 50, 60, 72, 73, 78, 79, 80, 102, 103, 130, 137, 153, 155, 201, 206, 267, 275, 300, 308, 365, 387, 431, 456, 459, 476, 482

BROMINE 411

BURIAL 3, 5, 18, 62, 88, 160, 167,

BURIAL 169, 173, 176, 185, 186, 187, 190, 191, 200, 211, 216, 260, 302, 307, 331, 330, 335, 342, 351, 352, 392, 402, 405, 463, 475, 516, 522, 556, 591, 613, 620, 622, 685, 681, 693, 695

BURIAL, INTERMEDIATE 176

BURIAL, SHALLOW 206, 277, 391

BURNUP 695, 698

*** C ***

CAESIUM 98, 330, 159

CAISSONS 205

CALCINATION 154, 204, 330, 346, 358, 392, 405, 577, 580, 583, 587, 602, 609, 615, 625, 640, 646, 648, 649, 660, 671, 680, 684, 693, 696, 702

CALCITE 196, 233, 286, 282, 287, 588, 596, 599, 602, 604, 615, 629, 650, 660, 665, 671, 695

CALCITE 183, 397

CALCIUM 29, 58, 88, 169, 286, 293, 290, 373, 576, 578, 630, 655

CALCIUM CARBONATE 58, 403

CALCIUM CARBONATE CONTENT 397

CALCIUM COMPOUNDS 85, 180

CALCULATIONS 130, 238, 271, 286, 363, 512, 585

CALICHE 196

CALIFORNIUM 209 87

CALIFORNIUM 252 133, 134

CANCRINITE 575

CANISTERS 29, 81, 82, 102, 109, 110, 127, 132, 138, 148, 151, 157, 158, 367, 372, 481, 529, 530, 531, 581, 585, 550, 561, 583, 631, 685

CAP ROCK 353, 811, 812, 832, 838, 861

CAPPING 205, 206

CARBON 127, 166, 167, 397

CARBON DIOXIDE 234, 245

CARBON 13 397, 399

CARBON 14 238, 245, 259, 327, 397, 398, 404, 501, 690

CARCINOGENESIS 512

CARCINOMAS 309

CASKS 87, 90, 97, 110, 112, 181, 186, 152, 158, 160, 205, 112, 113, 307, 506, 530, 531, 568, 565, 568, 631, 651

CATION ADSORPTION EQUILIBRIUM CONSTANT 2

CATION CONCENTRATION 85, 293

CATION EXCHANGE CAPACITY 28, 29, 60, 217, 289, 409, 576, 675, 680

CATIONS 2, 27, 165, 170, 183, 191, 197, 217, 269

CAVES 161, 536

CAVITIES 5, 17, 34, 177, 273, 329, 348, 350, 359, 360, 368, 375, 377, 379, 391, 407, 429, 462, 490, 540

CELLS 310

CELLULOSE 91

CELSIAN 185

CEMENTS 87, 59, 60, 62, 121, 156, 213, 332, 467, 509, 536, 570, 571, 575, 577, 593, 595, 621, 622, 624, 636, 639, 641, 653, 662, 663, 668, 673, 678, 691, 693

CERAMICS 58, 109, 108, 115, 122, 128, 131, 151, 155, 306, 627, 635, 636, 642, 640, 663, 665, 690, 696, 689

CESIUM 101, 261

CESIUM 103 28

CESIUM 108 227, 590, 590, 623, 624

CESIUM 242 133

CERIUM 569

CESIUM 18, 33, 38, 58, 68, 101, 105, 113, 155, 157, 165, 168, 218, 269, 281, 282, 292, 299, 302, 303, 332, 336, 380, 391, 420, 487, 467, 575, 582, 590, 615, 619, 629, 638, 689, 659, 660, 664, 668, 675, 683, 687, 688

CESIUM 134 239, 297, 317, 500, 501, 590

CESIUM 135 264

CESIUM 137 28, 169, 213, 230, 238, 261, 283, 285, 387, 377, 468, 486, 500, 501, 504, 620, 639, 655

CHALKS 68

CHELATES 7, 8, 9, 220, 255, 288, 486

CHEMICAL ANALYSIS 8, 6, 32, 39, 65, 166, 167, 168, 180, 377, 387, 388, 468, 575, 598, 600

CHEMICAL COMPOSITION 58, 79, 138, 155, 204, 213, 383, 428, 485, 602, 620, 628, 656, 679

CHEMICAL PROPERTIES 3, 6, 15, 16, 27, 31, 32, 42, 58, 66, 70, 77, 78, 80, 81, 89, 93, 96, 102, 108, 105, 108, 111, 115, 122, 132, 145, 149, 151, 155, 157, 204, 220, 230, 231, 256, 288, 308, 310, 360, 372, 375, 427, 482, 468, 467, 477, 480, 536, 627, 635, 637, 660, 668, 672, 679, 680, 693, 687, 691

CHEMICAL REACTIONS 7, 8

CHEMISTRY 16, 79, 81, 102, 132, 151, 157, 231, 637, 668

CHILDREN 238, 500

CHIPMUNKS 289

CHLORIDES 160, 387, 377, 397

CHLORINE 12, 42, 45, 180, 416

CHLORITES 38, 87, 437, 493

CHROMIUM 68, 84

CHROMIUM COMPOUNDS 680

CLADDING 97, 120, 125, 127, 205, 315, 332, 333, 416, 496, 529, 541, 545, 550, 625, 633, 681

CLASSIFICATION 120, 696

CLAYS 10, 33, 38, 39, 52, 60, 66, 74, 165, 173, 178, 183, 196, 197, 205, 210, 213, 227, 241, 287, 255, 265, 269, 288, 291, 297, 302, 308, 305, 342, 347, 365, 376, 388, 392, 403, 413, 423, 425, 426, 430, 458, 460, 463, 471, 493, 536, 575, 582, 609, 617, 641, 677, 679, 680, 689

CLEARTE 559

CLINOPTILOLITE 572, 639, 646, 649, 655, 677

COBALT 281, 303, 588

COBALT 60 168, 285, 584, 639

COLLOIDS 32, 92, 217, 288, 381, 696

COMBUSTION 309, 583, 599, 625, 687

COMMUNITIES 520

COMPACTION 52, 66, 269, 309, 381, 342, 595, 663, 702

COMPLEXES 7, 8, 45, 78, 203, 217, 269, 288, 310, 416, 468, 686

COMPUTER CODES 10, 25, 26, 38, 49, 67, 85, 105, 232, 238, 239, 282, 259, 267, 273, 291, 369, 390, 530, 560, 568, 651, 690, 691, 698, 702

COMPUTER PROGRAMS 20, 26, 55, 56, 100, 178, 197, 215, 216, 219, 222, 223, 232, 241, 250, 256, 262, 267, 269, 275, 279, 308, 373, 462, 466, 472, 485, 585, 582, 591, 631, 651, 667, 668, 691

COMPUTERS 161, 222, 237, 252, 280, 376, 383

CONCENTRATION 6, 173, 179, 218, 267, 276, 277, 291, 397, 523, 588, 600, 603, 607, 670

CONCENTRATION FACTORS 257, 261, 281, 405, 699

CONCRETES 59, 60, 95, 99, 100, 121, 125, 199, 206, 312, 329, 362, 386, 387, 389, 467, 536, 598, 622, 624, 625, 653, 655, 668, 673, 678, 682, 685, 698

CONDENSATION 160

CONSTRUCTION 296

CONTAINER INTEGRITY 92, 95, 97, 107, 126, 130, 133, 186, 159, 161, 169, 176, 199, 368, 471, 516, 556, 587, 631, 669

CONTAINERS 18, 20, 42, 47, 56, 67, 80, 85, 89, 92, 98, 95, 97, 101, 103, 110, 116, 119, 120, 121,

KEYWORD INDEX

CONTAINERS 123, 125, 127, 128, 129, 130, 132, 133, 136, 137, 140, 141, 142, 143, 146, 147, 150, 152, 158, 159, 160, 184, 196, 205, 206, 208, 210, 260, 261, 260, 291, 296, 309, 312, 329, 336, 341, 348, 346, 356, 359, 372, 382, 383, 396, 421, 433, 463, 471, 473, 491, 496, 501, 509, 552, 553, 557, 558, 573, 587, 612, 620, 629, 660, 665, 669, 671, 685

CONTAINMENT 30, 62, 97, 98, 99, 100, 101, 108, 123, 146, 150, 151, 157, 170, 179, 187, 190, 191, 192, 197, 200, 203, 216, 256, 281, 263, 260, 292, 295, 304, 317, 331, 336, 339, 351, 356, 362, 366, 368, 372, 373, 402, 413, 405, 452, 457, 465, 471, 477, 511, 518, 522, 523, 526, 551, 552, 558, 581, 610, 613, 630, 637, 642, 643, 640, 670, 698

CONTAINMENT TRANSPORT 267, 640, 668

CONTAINMENTS 97

CONTAMINATION 83, 85, 91, 165, 179, 181, 182, 191, 192, 199, 203, 201, 208, 215, 221, 292, 295, 305, 307, 308, 331, 336, 339, 366, 415, 489, 561, 506, 567, 581, 582, 610, 692, 693

CONVECTION 60, 187, 262, 273, 278, 275, 289, 308

COOLANTS 96, 107, 107, 123, 125, 140, 150, 200, 325, 332, 563, 573, 624

COPPER 132, 159

CORES 28, 69, 72, 87, 90, 170, 196, 281, 300, 312, 332, 358, 387, 358, 389, 390, 396, 411, 412, 422, 425, 426, 479

CORROSION 12, 80, 102, 103, 116, 117, 129, 130, 132, 137, 135, 141, 148, 151, 159, 336, 361, 416, 591, 604, 631

CORROSION RATE 129

COST BENEFIT ANALYSIS 55, 103, 109, 136, 159, 169, 176, 205, 206, 217, 316, 322, 327, 306, 705, 366, 396, 433, 438, 448, 450, 473, 478, 475, 492, 496, 499, 500, 501, 518, 521, 522, 525, 586, 559, 566, 578, 593, 598, 610, 620, 632, 607, 652, 653, 672, 683, 695, 698

COST ESTIMATES 175, 605

COSTS 186, 206, 213, 346, 387, 388, 500, 502, 520, 527, 601, 636, 659, 685

CRACKING 129

CREEKS 201

CREEP 10, 20, 25, 73, 95, 178, 277, 312, 373, 386, 500

CRIBS 170, 189, 286, 391

CRITICAL MASS 157

CRITICAL PATHWAY 261, 260, 309, 339

CRITICALITY 176, 552

CROPS 255, 339, 369

CRUSTACEANS 261

CRYSTALLIZATION 100, 105, 111, 115, 122, 138, 605, 679

CURIUM 15, 157, 214, 255, 280, 301, 409, 420, 505, 590, 700

CURIUM 242 138, 310

CURIUM 243 310

CURIUM 244 87, 134, 300, 309, 310

CURIUM 245 264

*** D ***

DAYS 309

DAUGHTER PRODUCTS 257, 261, 309, 330, 357, 437

DECAY HEAT 17, 40, 50, 78, 848

DECLADDING 97, 336, 603

DECOMMISSIONING 169, 175, 211, 300, 309, 326, 327, 329, 333, 341, 344, 457, 514, 515, 529, 556, 693

DECONTAMINATION 58, 60, 175, 197, 199, 211, 265, 296, 308, 309, 326, 333, 338, 341, 416, 556, 572, 578, 588, 592, 593, 613, 617, 620, 625, 633, 646, 647, 648, 649, 672, 680, 681, 687, 692, 693, 697

DECONTAMINATION FACTORS 351, 409, 576, 583, 587, 618, 649

DECREPITATION 72

DEEP 271

DEFORMATION 10, 12, 20, 51, 55, 143, 178, 237, 361, 373, 450, 466

DEGRADATION 7, 8, 9, 46, 47, 107, 135, 160, 445, 670

DENITRATION 625, 660

DENSIFICATION 569

DENSITY 53, 59, 60, 61, 66, 73, 77, 204, 262, 263, 269, 291, 302, 305, 437, 475, 535, 597, 602, 683

DENUDATION 402, 425

DEPOSITION 255

DEPTH 26, 38, 262, 263, 291

DESERTS 305

DESIGN 12, 18, 20, 51, 55, 56, 75, 85, 92, 98, 97, 107, 109, 112, 119, 127, 136, 143, 150, 172, 178, 205, 263, 312, 313, 316, 326, 328, 333, 302, 384, 361, 372, 381, 388, 387, 388, 389, 390, 421, 430, 440, 457, 466, 473, 482, 491, 492, 503, 503, 529, 538, 531, 533, 538, 536, 537, 539, 541, 543, 545, 546, 567, 568, 569, 550, 551, 552, 553, 554, 555, 557, 559, 561, 563, 570, 586, 609, 630, 681, 683, 651, 666, 670, 679, 697

DESORPTION 28, 32, 216, 299

DETERGENTS 626

DETERIOR 397

DEVITRIFICATION 620, 659, 679

DIAGENESIS 87, 410

DIAPIRS 5, 38, 161, 413, 430, 450

DICKITE 575

DIFFRACTION, X-RAY 27, 33, 115

DIFFUSION 62, 88, 118, 120, 128, 131, 188, 215, 234, 236, 239, 252, 258, 263, 266, 267, 273, 278, 279, 286, 292, 301, 303, 311, 335, 376, 382, 415, 522, 536, 591, 604, 639

DIFFUSION COEFFICIENT 218

DILUTION 311, 339, 522

DISCHARGE 206, 690

DISPERSION 123, 173, 278, 286, 289, 302, 311, 369, 413

DISPERSION COEFFICIENT 226, 232, 260, 276, 300, 303

DISPERSIVITY 215, 291

DISPOSAL FACILITIES 277, 690

DISPOSAL SITE 6, 7, 8, 9, 12, 55, 58, 98, 109 161, 162, 163, 168, 169, 179, 181, 182, 190, 191, 192, 193, 195, 197, 198, 199, 200, 201, 203, 205, 206, 207, 210, 216, 222, 265, 268, 290, 295, 300, 307, 316, 321, 324, 326, 328, 331, 339, 384, 351, 366, 368, 373, 402, 415, 428, 445, 457, 460, 463, 465, 471, 472, 487, 500, 508, 515, 516, 522, 530, 531, 533, 539, 555, 673

DISPOSAL, STACK 172

DISSOLUTION 5, 9, 42, 80, 263, 348, 353, 407, 412, 425, 472, 438, 439, 456, 459, 460, 461, 462, 478, 482, 536, 540, 647

DISTILLATION 609

DISTRIBUTION 208, 270, 285, 377, 489

DISTRIBUTION COEFFICIENT 1, 28, 32, 85, 91, 183, 217, 219, 261, 262, 269, 284, 288, 289, 291, 293, 295, 302, 350, 407, 576, 600

DIVERSION 496

DOLERITES 219

DOLOMITE 38, 39, 284, 300, 390, 411, 430, 431, 483

DOSE COMMITMENTS 118, 223, 233, 238, 260, 271, 281, 308, 309, 318, 339, 346, 500, 501, 521, 565, 652

DOSE RATE 42, 67, 81, 99, 123, 260, 271, 309, 339, 363, 477, 533, 545, 610, 633

DCSRS 62

DOSIMETERS 12, 203

DRAINAGE 100, 181, 189, 265

DRAINAGE CHARACTERISTICS 176, 189, 420

DRINDONS 242

DRILLING 34, 264, 312, 329, 354, 372, 389, 390, 396, 398, 422, 478, 478, 479, 482, 536, 543, 546

DRINKING WATER 84, 261, 264, 309, 587

DRUGS 5, 47, 91, 106, 136, 142, 146, 184, 185, 186, 581, 615, 676, 553, 598, 607, 612, 635

DRYING 468, 687, 622, 625

DUSTS 296, 300

DYES 311

*** E ***

EARTHQUAKES 176, 268, 389, 481, 425, 444, 455, 480, 512, 552

ECOLOGY 268, 339

ECONOMICS 13, 103, 125, 147, 152, 156, 185, 187, 287, 259, 308, 312, 318, 321, 322, 324, 325, 333, 359, 360, 368, 372, 405, 413, 422, 480, 498, 506, 507, 509, 510, 516, 518, 519, 520, 525, 526, 546, 573, 592, 605

ECOSYSTEMS, AQUATIC 258, 261, 264, 337, 339, 367

ECOSYSTEMS, TERRESTRIAL 258, 256, 260, 339

EDTA 8

EFFLUENTS 192, 238, 457, 501, 521, 550, 555, 652, 671, 704

EFFLUENTS, AIRBORNE 172, 296, 363, 521, 691, 656

EFFLUENTS, CHEMICAL 521

EFFLUENTS, LIQUID 296, 521, 652, 691

EH 11, 15, 70, 99, 219, 220, 229, 269, 397

ELASTOPLASTICITY 20, 67, 86, 140, 183, 178, 380, 425, 480

ELECTROCHEMISTRY 129, 168

ELECTRODEPOSITION 159, 600

ELECTROSTATICS 616

ELECTRON MICROPHONE 115

EMISSIONS 580, 590, 600, 700

ENCAPSULATION 17, 40, 60, 121, 126, 131, 151, 335, 336, 387, 531, 569, 572, 576, 583, 592, 593, 605, 606, 610, 625, 627, 629, 635, 638, 646, 649, 663, 665, 675, 676

ENRICHMENT 520

ENVIRONMENT 152, 179, 187, 190, 250, 259, 317, 324, 331, 337, 339, 345, 351, 353, 402, 463, 504, 509, 511, 514, 515, 522, 527, 530, 555, 556, 558, 610

ENVIRONMENTAL EXPOSURE PATHWAY 123, 233, 235, 254, 256, 260, 270, 277, 279, 281, 306, 308, 309, 310, 324, 325, 326, 339, 340, 394, 449, 490, 526, 633

ENVIRONMENTAL IMPACT STATEMENTS 259, 313, 320, 457, 482

ENVIRONMENTAL IMPACTS 335, 538

EQUATIONS 2, 10, 20, 28, 41, 42, 49, 55, 61, 62, 69, 90, 110, 119, 138, 143, 165, 178, 197, 215, 239, 241, 243, 256, 257, 258, 262, 264, 265, 266, 267, 272, 284, 286, 289, 291, 292, 294, 295, 303, 302, 303, 311, 354, 376, 398, 428, 462, 472, 485, 538, 576, 581, 601, 626, 632, 639, 672, 676

EQUILIBRIUM CONSTANT 29, 217, 572, 606, 677

EQUIPMENT 63, 97, 106, 109, 127, 130, 150, 172, 175, 182, 186, 212, 297, 313, 322, 326, 355, 361, 361, 490, 528, 531, 538, 539, 543, 547, 548, 549, 556, 557, 567, 570, 603, 607, 636, 638, 628, 630, 638, 641, 651, 658, 670, 672, 695, 697

EROSION 264, 268, 277, 300, 309, 420, 425, 428, 456

ESTUARIES 209, 339

EUROPIUM 7, 8, 157, 228

EUROPIUM 152 28

EUROPIUM 154 590, 628

EVALUATION 51, 53, 52, 55, 66, 68, 84, 107, 109, 129, 175, 217, 265, 255, 309, 316, 331, 340, 345, 346, 366, 383, 498, 505, 512, 530, 531, 539, 623, 651, 653

EVAPORATION 107, 337, 338, 359, 360, 456, 523, 583, 592, 598, 603, 613, 615, 617, 628, 625, 666, 680, 681, 693, 698

EVAPORATION RATE 603

EVAPORATOR CONCENTRATE 578, 596, 603, 612

EVAPORATORS 592, 603, 612, 633, 666, 667

EVAPORITES 16, 73, 78, 79, 281, 263, 330, 378, 385, 407, 411, 412, 413, 417, 425, 431, 432, 438, 439, 442, 453, 458, 456, 459, 482

EXCAVATION 18, 28, 22, 67, 85, 173, 178, 186, 277, 296, 389, 350, 360, 382, 388, 390, 396, 413, 422, 425, 440, 580, 581, 583, 552

EXHAUST GASES 197, 300, 309, 681

EXPANSION 256

EXPERIMENTS 16, 54, 72, 78, 102, 155

EXPLOSIONS 309, 336, 350

EXPLOSIONS, NON-NUCLEAR 343, 543

EXPLOSIONS, NUCLEAR 163, 343, 363, 377

EXPOSURE, EXTERNAL 209, 230, 281, 309, 363, 581

EXPOSURE, INTERNAL 209, 238, 254, 281, 363, 580, 581

EXPOSURE, OCCUPATIONAL 99, 206, 209, 329, 340, 383, 493, 552, 651

EXPOSURE, POPULATION 206, 254, 261, 264, 308, 309, 493, 490, 581, 586, 512, 530

EXTRACTION 280, 336, 416, 580

*** F ***

FABRICATION 192, 573

FALLOUT 230, 235, 261, 306, 308

FARMS 78, 226, 232, 269, 277, 318, 368, 379, 380, 386, 399, 410, 451, 452, 453, 458, 430, 439, 452, 455, 480, 481, 484, 489

FELDSPARS 87, 109, 418

FIELD STUDIES 12, 19, 36, 56, 61, 71, 87, 91, 99, 100, 109, 120, 129, 137, 161, 162, 170, 173, 179, 182, 184, 189, 191, 195, 198, 201, 213, 214, 227, 235, 271, 276, 283, 286, 290, 291, 306, 312, 313, 316, 320, 333, 335, 348, 350, 356, 353, 359, 360, 363, 366, 370, 371, 375, 381, 382, 384, 387, 388, 389, 390, 393, 397, 398, 399, 400, 403, 406, 408, 410, 411, 415, 417, 418, 419, 420, 421, 422, 426, 428, 430, 438, 439, 446, 446, 447, 453, 458, 455, 460, 461, 470, 472, 478, 479, 482, 484, 489, 530, 531, 534, 535, 536, 539, 567, 581, 582, 630, 605, 649

FILTERS 182, 498, 573, 595, 616, 628, 633, 697, 702

FILTRATION 269, 416

FIRE HAZARDS 110, 202, 296, 560

FIRES 260, 309, 343, 433, 512, 552

FISH 201, 261

FISSION 133, 236, 650

FISSION PRODUCTS 1, 32, 50, 100, 122, 128, 140, 157, 159, 170, 179, 191, 199, 220, 233, 236, 287, 258, 259, 259, 261, 260, 272, 288, 291, 331, 347, 358, 375, 402, 409, 433, 434, 506, 528, 536, 569, 570, 577, 581, 585, 586, 599, 601, 604, 607, 609, 610, 613, 617, 625, 631, 637, 642, 643, 648, 605, 606, 678, 680, 681, 690, 702

FISSURES 128, 209, 252, 287

FIXATION 5, 10, 50, 58, 199, 206, 218, 220, 230, 247, 259, 292, 347, 388, 389, 390, 405, 406, 564, 570, 582, 593, 605, 629,

KEYWORD INDEX

VIBRATION 637, 644, 646, 658, 675, 679, 683, 685

VISCOSITIES 318, 593, 613

VISCOSITY 256, 260

VISCOUS 307, 309, 498

VISCOUSES 307

VITA BUTT 87, 213, 670

VOID INCLUSIONS 36, 72

VOLATILE ORGANICS 49, 80, 215, 239, 242, 267

VOLCANICITY 309, 503, 646

VOLTAMMETRY 200

VOLUMETRY 696

WAY BOW 59, 601

WORK 256

WORK CHAIRS 85, 235, 238, 254, 261, 268, 280

WORKS 180, 200, 201, 203

WRITTEN REVIEW 58

YEARBOOKS 28, 34, 42, 48, 56, 85, 98, 179, 210, 218, 242, 252, 289, 292, 329, 350, 360, 372, 375, 387, 388, 389, 398, 399, 396, 397, 590, 598, 601, 608, 625, 637, 638, 643, 647, 669, 676, 680, 681, 682, 688, 689, 505, 535, 536, 543

YELLOWSTONE SYSTEMS 275, 311, 348, 366

YOUNG 614

ZONE CYCLES 125, 225, 226, 251, 321, 322, 324, 325, 326, 327, 338, 541, 535, 577, 550, 552, 590, 678, 682, 690, 700

ZONE ELEMENTS 121, 159, 169, 332, 506, 522, 573, 579, 582, 631, 687

ZONE FABRICATION 521, 652

ZONE REFRACTORY 35, 97, 125, 150, 189, 233, 317, 318, 322, 336, 416, 487, 510, 519, 528, 533, 537, 580, 583, 587, 588, 607, 611, 631, 633, 637, 662, 666, 681, 688, 696, 698, 705

ZONE ROOM 96, 560, 629, 698

ZONES 97, 100, 521, 566, 690

ZONES 310

ooo G ooo

GABBROS 40, 299, 372

GAMMOLITHOS 39

GASES 8, 9, 43, 46, 47, 57, 72, 78, 91, 98, 100, 127, 202, 230, 238, 265, 296, 327, 330, 381, 387, 388, 395, 359, 360, 413, 425, 430, 500, 506, 577, 580, 590, 628, 680, 690, 695

GASTROINTESTINAL TRACT 230, 500,

GASTROINTESTINAL TRACT 501

GENETICS 309, 512

GEOCHEMISTRY 35, 36, 54, 65, 70, 97, 105, 137, 149, 157, 171, 190, 217, 218, 254, 269, 286, 308, 312, 317, 334, 375, 397, 406, 428, 435, 447, 451, 452, 485, 487

GEOGRAPHY 207, 420, 428

GEOLOGIC DEPOSITS 15, 46, 51, 84, 220, 229, 247

GEOLOGIC FORMATIONS 72, 78, 84, 102, 192, 193, 194, 195, 203, 219, 256, 313, 319, 327, 364, 368, 390, 399, 400, 401, 408, 413, 425, 428, 429, 443, 458, 460, 484, 492, 528, 585, 558, 55-

GEOLOGIC HISTORY 264, 383, 390, 412, 417, 425, 432, 438, 439, 453, 485

GEOLOGIC MAPPING 193, 194, 195, 365, 413, 387, 388, 389, 7-8, 394

GEOLOGIC SERIES 38, 76, 183, 194, 195, 225, 237, 239, 351, 365, 387, 404, 425, 445, 450, 458, 460, 464, 470, 485

GEOLOGIC STRUCTURES 38, 373, 388, 389, 389, 397, 398, 399, 391, 393, 399, 400, 401, 403, 406, 412, 417, 420, 427, 428, 429, 430, 432, 438, 439, 440, 446, 453, 454, 455, 458, 466, 480, 482, 485

GEOLOGY 5, 17, 26, 68, 77, 97, 157, 161, 163, 171, 188, 189, 191, 203, 207, 216, 229, 260, 291, 299, 304, 312, 317, 318, 319, 322, 328, 329, 338, 339, 335, 339, 348, 351, 353, 362, 364, 365, 366, 368, 369, 370, 371, 373, 375, 378, 379, 380, 387, 388, 389, 390, 391, 393, 394, 399, 400, 401, 402, 403, 406, 408, 412, 417, 420, 426, 427, 428, 429, 431, 432, 435, 439, 443, 444, 451, 452, 453, 454, 458, 459, 461, 463, 464, 466, 471, 472, 479, 480, 483, 485, 487, 488, 503, 546, 522, 552, 555, 558, 560, 581, 610, 686, 681

GEOPHYSICAL SURVEYS 12, 13, 316, 328, 365, 366, 371, 387, 388, 389, 390, 418, 422, 426, 428, 435, 437, 454, 470, 472, 485, 533

GIBBSITE 78

GLACIERS 351, 352, 425, 441, 470, 491, 492

GLASS 1, 30, 38, 48, 54, 88, 93, 95, 99, 100, 109, 111, 115, 118, 121, 123, 128, 126, 128, 131, 133, 138, 150, 151, 153, 155, 233, 246, 259, 282, 315, 330, 332, 346, 347, 358, 369, 372, 375, 380, 384, 392, 405, 407, 478, 490, 573, 580, 585, 587, 591, 602, 604, 605, 607, 608, 616, 620, 627, 635, 643, 644, 649, 656, 657, 658, 659, 661, 663, 665, 668, 669, 679, 683, 684, 685, 686, 689, 698

GLAUCONITE 290, 364

GLOVE BOXES 309, 622, 672

GLOVES 83

GNEISSES 68, 276

GOLD 159

GRABITES 11, 15, 21, 28, 29, 56, 63, 68, 69, 75, 77, 83, 149, 150, 151, 219, 220, 237, 268, 276, 313, 318, 322, 329, 372, 373, 378, 381, 383, 385, 397, 398, 404, 412, 425, 427, 437, 442, 447, 458, 469, 478, 481, 487, 535, 536, 546, 547, 552, 553, 554, 575

GRANODIORITES 63

GRAPHITES 694, 695

GRASSES 258

GRAVELS 173, 183, 192, 385, 388, 438, 536

GROUND WATER 6, 11, 17, 19, 28, 32, 38, 48, 73, 74, 88, 89, 143, 159, 156, 157, 188, 189, 192, 193, 194, 195, 197, 198, 200, 203, 215, 216, 217, 218, 219, 221, 232, 234, 237, 240, 241, 243, 256, 262, 263, 264, 265, 268, 269, 277, 279, 282, 284, 293, 295, 296, 300, 301, 302, 306, 307, 308, 309, 328, 335, 335, 351, 352, 353, 365, 368, 372, 375, 376, 377, 378, 387, 388, 389, 390, 397, 398, 402, 407, 418, 419, 422, 423, 425, 429, 431, 432, 437, 438, 440, 445, 447, 459, 460, 461, 470, 471, 472, 489, 535, 536, 540, 546, 547, 548, 549, 552, 553, 581, 605

GROUTING 210, 213, 214, 364, 391, 394, 395, 401, 404, 429, 443, 536, 639, 641, 663

GYPSUM 613

ooo H ooo

HALF-LIFE, BIOLOGICAL 207, 339, 699

HALF-LIFE, RADIOLOGICAL 15, 197, 339, 618

HALITE 16, 73, 74, 82, 161, 281, 263, 384, 407, 411, 412, 432, 438, 478, 482

HALLOYSITE 575

HALOGENS 588

HEAVY ANALYSIS 5, 11, 35, 42, 44, 92, 118, 120, 123, 167, 152, 186, 206, 211, 221, 223, 250, 258, 261, 264, 278, 308, 309, 312, 313, 315, 316, 317, 319, 327, 330, 335, 336, 338, 380, 383, 384, 385, 386, 387, 389, 402, 407, 438, 457, 478, 475, 493, 498, 501, 505, 506, 512, 515, 517, 526, 528, 542, 552, 558, 557, 560, 562, 564, 565, 567, 585, 6-7, 696, 699, 703

HEALTH 169, 231, 522, 528, 610, 698

HEAD PRESSURE 618

HEALTH PHYSICS 106

HEAT CAPACITY 76, 823

HEAT EXCHANGE 37, 58, 69, 96, 107, 2[illegible], 406, 560, 573

HEAT RESISTANCE 107, 119, 560

HEAT SOURCES 19, 20, 25, 37, 56, 67, 75, 85, 89, 282, 237, 370, 377, 381, 382, 386, 389, 390, 421, 425, 429, 486, 487, 491, 498, 530, 535, 573, 698, 702

HEAT TRANSFER 18, 25, 58, 57, 67, 75, 85, 89, 107, 119, 267, 372, 373, 385, 388, 381, 383, 390, 403, 411, 421, 486, 487, 532, 603, 632

HELIUM 131, 133, 180, 290, 450, 608

HEMING BONDS 320, 705

HOMOGENIZE 145

HORTION 171

HOT ROCKS 17, 50, 78, 157, 262, 375, 468

HOT CELLS 735, 687, 689

HOLES 295, 782

HOLIC ACIDS 380

HOMOGENITY 456

HYDRAULIC CONDUCTIVITY 68, 98, 290, 302, 376, 377, 610, 688

HYDRAULIC FLOW 456

HYDRAULIC FRACTURING 69, 213, 355, 366, 391, 399, 401, 409, 425, 487, 608

HYDRAULIC GRADIENT 239, 237, 267, 302, 376, 608

HYDRAULIC HEAD 98, 389, 419, 456

HYDRAULIC VELOCITY 37, 81, 232, 363, 267, 375, 376, 456

HYDRAULICS 85, 268, 377, 398, 410, 425

HYDRODYNAMICS 192, 203, 215, 223, 302, 243, 265, 266, 291, 395, 475, 489, 459, 468, 849

HYDROGEN 82, 83, 91, 99, 292, 479, 573, 577

HYDROLOGY 5, 17, 26, 68, 77, 87, 163, 180, 189, 191, 197, 200, 203, 207, 216, 219, 221, 222, 259, 312, 317, 319, 324, 334, 335, 339, 348, 351, 353, 357, 365, 366, 368, 371, 377, 380, 383, 388, 387, 388, 389, 390, 393, 399, 402, 406, 408, 412, 418, 420, 426, 427, 428, 429, 430, 431, 432, 435, 436, 438, 448, 449, 451, 452, 456, 459, 460, 461, 471, 472, 479, 487, 488, 522, 547, 548, 549, 552, 585, 610

HYDROLYSIS 1, 6, 98, 269, 288

HYDROTHERMAL ALTERATIONS 18, 19, 89, 189, 282, 387, 389, 390, 390

HYDROXIDES 6, 606

*** I ***

ICE 309, 352, 491, 492

IGNEOUS ROCKS 11, 15, 21, 23, 29, 31, 32, 40, 49, 61, 65, 68, 75, 132, 189, 150, 195, 218, 219, 249, 251, 252, 317, 318, 329, 372, 378, 375, 380, 381, 395, 397, 402, 406, 410, 413, 427, 427, 442, 444, 446, 447, 451, 453, 469, 470, 481, 488, 487, 489, 528, 546, 547, 549, 579

ILLITE 10, 22, 33, 39, 64, 87, 165, 270, 403, 483, 675

IMMOBILIZATION 17, 40, 45, 60, 62, 111, 181, 157, 211, 220, 250, 265, 309, 330, 391, 392, 397, 372, 463, 471, 476, 490, 490, 516, 570, 586, 587, 591, 596, 616, 621, 622, 626, 627, 630, 646, 655, 656, 657, 659, 663, 669, 670, 680, 681, 682

IMPACT SHOCK 92, 505

IMPLIMENTATION 309, 339, 341, 342, 383, 385, 389, 419, 465, 470, 490, 570, 571, 577, 580, 676, 622, 630, 650, 681, 682, 702

INERT GASES 691

INFILTRATION 162, 190, 292, 299, 581

INFORMATION 510, 526

INGESTION 238, 235, 250, 256, 261, 268, 271, 277, 303, 335, 560, 581, 703

INHALATION 237, 238, 250, 256, 277, 305, 325, 500, 581, 703

INJECTION 216, 368, 391, 390, 399, 400, 401, 419, 429, 490

INORGANIC COMPOUNDS 168

INSECTS 225

INSTRUMENTATION 196, 216, 658

INSTRUMENTS 6, 19, 56, 63, 572, 392, 335, 361, 382, 389, 390, 419, 437, 487, 530, 538, 535, 536, 539, 567, 618

INTRUSION 177

INVERTEBRATES 235

IODIDES 695

IODINE 32, 42, 127, 205, 207, 250, 291, 336, 702

IODINE 129 6, 238, 245, 259, 260, 269, 500, 501

IODINE 131 304, 238, 297, 501, 508, 620, 691

ION EXCHANGE 6, 27, 80, 85, 60, 104, 157, 179, 183, 191, 201, 250, 269, 272, 286, 300, 302, 327, 328, 339, 356, 366, 486, 503, 563, 595, 597, 600, 606, 609, 615, 621, 620, 636, 646, 667, 669, 668, 670, 672, 675, 676, 677, 693, 697, 688, 693

ION EXCHANGE CAPACITY 2, 27, 165, 183, 197, 217, 293, 525, 572, 597, 619, 638, 678, 690

IONIC PROCESSES 6, 32, 78, 67[illegible]

IONS 6, 16, 32, 39, 48, 165, 206, 292, 377, 576, 681

IRON 30, 168, 460, 569, 620, 656, 657

IRON COMPOUNDS 126, 658, 659

IRON 55 230

IRRADIATION 82, 87, 92, 90, 100, 602, 631, 605

IRRIGATION 222, 369

ISOTOPE RATIOS 87, 100, 631

ISOTOPES 40, 45, 65, 183, 383, 700

ISOTROPY 20, 398, 490, 535

*** J ***

JOINTS 65, 196, 218, 381, 413, 466, 465, 470, 481

*** K ***

KAOLIN 29, 575, 529, 689

KAOLINITE 70, 165, 197, 270, 288, 297, 336, 576, 675, 678

KINETICS 78, 82, 80, 655

KRYPTON 123, 127, 291

KRYPTON 85 181, 238, 285, 327, 355, 500, 591, 620

*** L ***

LABORATORY STUDIES 1, 2, 3, 4, 7, 8, 9, 11, 12, 13, 18, 16, 27, 28, 29, 30, 31, 32, 33, 36, 39, 39, 40, 42, 43, 44, 45, 46, 46, 47, 48, 51, 52, 53, 56, 57, 58, 59, 60, 63, 65, 65, 66, 69, 70, 71, 72, 73, 78, 76, 78, 79, 81, 82, 83, 86, 87, 88, 89, 90, 91, 95, 96, 99, 101, 102, 103, 104, 119, 113, 118, 122, 123, 126, 129, 129, 130, 131, 138, 139, 141, 145, 148, 152, 153, 155, 156, 160, 173, 175, 178, 182, 187, 198, 220, 220, 227, 229, 229, 230, 235, 240, 241, 246, 247, 248, 249, 251, 252, 255, 265, 271, 272, 278, 282, 284, 286, 288, 291, 292, 293, 298, 295, 297, 298, 299, 300, 301, 306, 306, 312, 313, 315, 316, 322, 325, 328, 329, 330, 358, 359, 360, 369, 375, 387, 388, 389, 390, 397, 401, 406, 409, 419, 423, 428, 437, 447, 464, 469, 481, 485, 532, 536, 560, 569, 570, 571, 572, 576, 574, 577, 578, 582, 589, 590, 597, 602, 604, 608, 605, 606, 607, 608, 617, 619, 623, 625, 628, 629, 630, 635, 636, 637, 638, 640, 641, 642, 649, 655, 656, 657, 658, 659, 661, 662, 664, 665, 668, 672, 673, 678, 679, 677, 678, 680, 686, 687, 689

LAKES 268, 299, 317

KEYWORD INDEX

LAKE BED 297, 300, 340, 344, 353, 417, 538

LABORATORIES 15, 271, 489

LABORATORY 58

LABORATORY 140 144

LASERS 681

LATTICE 33, 111, 113, 114, 126, 131, 178, 675

LAVA 355

LEACHING 3, 30, 44, 50, 60, 62, 85, 104, 118, 124, 128, 137, 150, 153, 155, 163, 186, 197, 218, 226, 228, 229, 237, 246, 256, 258, 261, 262, 264, 269, 276, 282, 302, 306, 309, 314, 315, 322, 326, 339, 356, 358, 365, 392, 425, 447, 463, 467, 476, 552, 569, 570, 575, 575, 581, 593, 596, 597, 602, 607, 608, 619, 620, 628, 636, 639, 643, 645, 647, 655, 659, 661, 663, 664, 665, 668, 669, 673, 676, 679, 680, 685

LEAD 65, 80, 132, 143, 159

LEAKS 144, 162, 186, 199, 222, 631

LEAVES 310

LEGISLATION 92, 125, 147, 165, 180, 322, 344, 495, 497, 508, 510, 511, 514, 515, 520, 527, 610

LEUCITE 145

LICENSING 92, 312, 319, 321, 355, 492, 495, 497, 500, 508, 512

LIGNITES 59

LIME 628

LIMESTONES 31, 50, 63, 68, 239, 289, 299, 300, 370, 371, 372, 393, 425, 427, 430, 431, 458, 520, 579, 602

LIMERS 101, 136, 192, 205

LITHIUM 160, 377, 600, 657

LITHIUM COMPOUNDS 658

LITHOLOGY 197, 365, 366, 379, 410, 417, 427, 483

LIVER 271

LIVESTOCK 334

LOAM 50, 200, 227, 280

LOESS 302

LOGGING 276, 536

LOGGING, CALIPER 437

LOGGING, ELECTRIC 417, 460, 461, 479

LOGGING, GAMMA 106, 411, 437, 479

LOGGING, NEUTRON 437

LOGGING, RESISTIVITY 437, 475

LOGGING, SONIC 437, 479

LOGGING, TEMPERATURE 437

LOGGING, WELL 189, 410, 430, 479

LUNGS 271, 309, 501

*** M ***

MAGNESIUM 160, 206, 377

MAGMAS 283

MAP 239, 253, 256, 261, 264, 271, 308, 309, 339, 340, 387, 500, 512, 526

MANGANESE 160, 658

MAPS 26, 189, 195, 216, 329, 365, 370, 411, 412, 430, 478, 479

MARINE 205

MARINE SYSTEMS 140, 230, 261, 289

MARLS 483

MASS BALANCE 29, 165, 572, 576

MATERIAL BALANCE 170

MAXIMUM PERMISSIBLE CONCENTRATION 62, 327, 291, 339, 508

MEASUREMENTS 16, 40, 61, 63, 76, 134, 172, 363, 364

MECHANICAL PROPERTIES 535, 536

MECHANICAL STRENGTH 665

MELTING 503

MERCURY 4, 84, 620

METAL 6, 101, 130, 132, 236, 306, [illegible], 614, 659, 663, 689

METAMORPHIC ROCKS 15, 68, 180, 219, 32[illegible], 375, 380, 402, 413, 453, 471, 488, 489

METEORITES 256, 260, 275, 389, 490

METEOROLOGY 207, 238, 259, 260, 268, 555, 652

METHODS 4, 7, 8, 16, 17, 26, 32, 40, 49, 61, 63, 65, 71, 72, 76, 78, 94, 97, 110, 115, 153, 172, 175, 182, 184, 186, 190, 203, 211, 213, 218, 225, 227, 284, 285, 286, 256, 257, 276, 318, 326, 333, 336, 350, 350, 375, 391, 399, 400, 422, 428, 436, 470, 477, 480, 481, 490, 512, 515, 502, 503, 507, 508, 509, 555, 561, 567, 569, 588, 593, 590, 597, 605, 607, 609, 611, 612, 614, 616, 618, 629, 636, 640, 641, 648, 650, 658, 665, 686

MICA 10, 488

MICE 283

MICROORGANISMS 7, 8, 9, 167, 228, 310

MICROSCOPY 115, 122

MILLING 326

MILLS 327, 339

MINERALOGY 16, 28, 52, 66, 77, 87, 105, 122, 410, 461, 480, 483

MINERALS 1, 29, 30, 38, 39, 54, 79, 80, 83, 105, 157, 236, 241, 250, 269, 286, 300, 307, 388, 390, 412, 417, 425, 428, 446, 469, 485, 606, 640, 653, 677

MINES 12, 13, 34, 51, 55, 55, 107, 177, 316, 320, 329, 339, 359, 360, 361, 370, 378, 381, 395, 461, 487, 489, 497, 530, 531, 533, 535, 589

MINING 28, 170, 316, 322, 326, 329, 342, 354, 395, 396, 417, 457, 455, 4[illegible], 536, 540, 583, 586, 585

MODELS 24, 25, 30, 37, 40, 50, 51, 56, 57, 67, 71, 85, 88, 112, 140, 147, 170, 188, 219, 222, 225, 226, 232, 234, 237, 252, 254, 260, 261, 263, 267, 269, 275, 279, 285, 302, 303, 311, 313, 315, 324, 330, 333, 369, 370, 393, 396, 397, 399, 422, 448, 466, 477, 498, 505, 506, 510, 517, 521, 532, 536, 540, 543, 562, 563, 566, 569, 646, 652, 691, 696

MODELS, MATHEMATICAL 10, 20, 24, 25, 37, 48, 57, 67, 85, 88, 100, 177, 188, 197, 215, 216, 217, 221, 222, 225, 226, 232, 239, 237, 242, 243, 256, 258, 260, 262, 266, 267, 273, 279, 284, 287, 289, 291, 303, 306, 312, 349, 369, 373, 374, 383, 386, 389, 422, 428, 436, 440, 462, 466, 477, 494, 521, 532, 540, 560, 562, 563, 566, 639, 652, 668, 678, 691

MOISTURE 13, 61, 91, 124, 173, 268, 288, 361, 449, 536

MOISTURE CONTENT 384, 437

MOLLUSKS 261

MOLYBDENUM 54, 128, 155, 608, 699

MONAZITE 17, 54, 600

MONITORING 106, 116, 162, 168, 171, 172, 178, 179, 182, 192, 198, 199, 200, 231, 205, 209, 210, 213, 218, 265, 268, 307, 308, 313, 320, 331, 332, 355, 375, 445, 455, 457, 470, 472, 493, 515, 526, 530, 536, 552, 553, 557, 558, 555, 556, 567, 583, 590, 600, 628, 631, 685, 650, 693

MONTMORILLONITE 27, 32, 64, 66, 70, 165, 183, 197, 270, 403, 576, 675, 677, 678

MORDENITE 659

MORTALITY 253

MUDSTONES 68

MUSCLE 271

MUTATIONS 304

*** N ***

NATURAL GAS 429

NEODYMIUM 54

NEPHELINE 685, 659

KEYWORD INDEX

NEPTUNIUM 11, 15, 32, 153, 220, 228, 250, 294, 299, 301, 424, 585, 600

NEPTUNIUM COMPOUNDS 159

NEPTUNIUM 237 47, 264, 293, 310, 487

NEPTUNIUM 239 590

NEUTRONS 138, 590, 650

NICKEL 148, 569, 656, 699

NICKEL 63 69

NICRIPS 95 285

NITRATES 60, 129, 168, 227, 612, 615, 640, 650

NITRIC ACID 628

NITRITES 139 168, 615

NITROGEN COMPOUNDS 98, 129, 363, 607

NITROGEN OXIDES 285, 695

NITROGEN 14 650

NUCLEAR FACILITIES 211, 245, 253, 256, 261, 280, 290, 320, 326, 327, 339, 400, 589, 628, 653, 691 692, 693, 696

NUCLEAR MATERIALS 120, 339, 496

NUCLEAR POWER 1, 253, 320, 322, 331, 516, 520, 528, 610, 690, 705

NUCLEAR POWER PLANTS 704

NUMERICAL METHODS 49, 215, 239, 282, 287, 276, 273, 275, 346, 462,

*** O ***

OCEANOGRAPHY 188, 303, 357, 415, 489

OCEANS 289

OIL 43

ORE DEPOSITS 65, 84, 379, 485

ORES 84, 520

ORGANIC COMPOUNDS 47, 87, 89, 100, 166, 167, 202, 217, 238, 310, 424, 486, 574, 577, 596, 678

OUTCROPS 87, 391, 830, 870

OVERBURDEN 366

OVERPACKING 33, 206, 383, 561

OXIDATION 12, 70, 103, 180, 345, 357

OXIDES 6, 30, 38, 108, 133, 236, 290, 110, 620, 636, 659

OXYGEN 98, 113, 126, 188

OXYGEN 16 398

OXYGEN 17 698

OXYGEN 18 357, 398

OZONE 363

*** P ***

PACKAGING 125, 130, 135, 136, 152, 186, 197, 212, 309, 326, 341, 342, 383, 433, 434, 550, 599, 613, 651

PARTICLE SIZE 28, 33, 44, 204, 302, 570, 584

PARTICLES 218, 230, 289, 290, 301, 310, 347, 377

PARTICLES, AIRBORNE 94, 254, 300, 369, 584

PARTITIONING 233, 297, 299, 424, 517, 579, 601

PELLETS 679

PERCHED WATER 192

PERCOLATION 173

PERMEABILITY 28, 29, 36, 48, 49, 52, 57, 59, 66, 69, 73, 77, 90, 132, 156, 173, 242, 262, 263, 267, 287, 312, 329, 364, 366, 368 389, 391, 398, 400, 408, 413 417, 421, 425, 429, 437, 486, 487, 470, 480, 487, 505, 535, 536

PERMEABILITY, INTRINSIC 97

PEROVSKITE 385

PERSONNEL 186, 209, 307, 343, 385, 510

PETROGRAPHY 204

PETROLEUM 334

pH 6, 11, 15, 28, 29, 48, 64, 88, 89, 93, 103, 132, 180, 183, 197, 217, 219, 220, 269, 286, 298, 302, 397, 409, 578, 591, 612, 623

PHENOLS 96

PHLOGOPITE 18

PHOSPHATE GLASS 108

PHOSPHATES 357, 489

PHOSPHORIC ACID 628

PHOTOCHEMISTRY 601

PHOTONS 601

PHYSICAL FORM 310, 386

PHYSICAL PROPERTIES 3, 16, 21, 22, 23, 24, 36, 52, 66, 72, 73, 76, 78, 79, 80, 81, 89, 108, 115, 122, 143, 156, 157, 204, 231, 256, 304, 375, 380, 384, 387, 388, 389, 390, 425, 427, 437, 442, 449, 460, 466, 469, 477, 480, 489, 505, 535, 536, 570, 605, 604, 638, 635, 659, 662, 669, 671, 679, 687, 691

PIEZOMETRIC SURFACE 192, 860

PIPES 107

PITS 164, 186, 196, 204, 307, 419, 478, 617, 630, 692, 693

PLANKTON 261, 311, 337

PLANTS, FUEL REPROCESSING 205, 702

PLANTS, WASTE TREATMENT 169, 172, 321, 324, 336, 349, 508, 520, 527, 583, 592, 593, 594, 612, 625, 632, 649, 682, 692, 693

PLASTICITY 12, 173, 359, 360, 403

PLASTICS 46, 136, 596

PLATINUM 133

PLUGGING 356

PLUTONIUM 8, 9, 15, 31, 32, 60, 70, 92, 106, 125, 153, 157, 159, 160, 170, 181, 188, 201, 230, 218, 216, 220, 224, 230, 235, 247, 248, 249, 250, 252, 255, 260, 270, 288, 293, 295, 299, 302, 303, 305, 317, 318, 324, 325, 332, 337, 356, 420, 447, 467, 476, 486, 490, 510, 523, 552, 571, 578, 582, 585, 588, 590, 599, 600, 601, 608, 610, 622, 623, 648, 656, 660, 672, 682, 688, 700, 702

PLUTONIUM COMPOUNDS 608, 707

PLUTONIUM ISOTOPES 252

PLUTONIUM 238 87, 87, 91, 68, 208, 271, 307, 308, 309, 310, 468, 700

PLUTONIUM 239 7, 43, 47, 91, 136, 144, 168, 208, 254, 256, 261, 269, 271, 290, 307, 308, 309, 310, 468, 700

PLUTONIUM 240 208, 256, 271, 308, 309, 700

PLUTONIUM 241 208, 264, 308, 309, 700

PLUTONIUM 242 208, 308, 309

PLUTONS 318, 469

POISONS 138

POLLUCITE 105, 113, 149, 282, 388, 619

POLLUTION, AIR 182, 277, 332, 339

POLLUTION, SOIL 45, 339

POLLUTION, WATER 189, 277, 339, 367, 455

POLONIUM 235

POLYETHYLENE 43, 91

POLYHALITE 78, 281, 425

POLYMERS 136, 216, 217, 249, 252, 369, 486, 592

POLYVINYL CHLORIDE 83

PONDS 320, 339

POPULATIONS 207, 238, 253, 317, 346, 347, 500, 511, 527, 565

POROSITY 28, 29, 49, 59, 60, 61, 69, 73, 191, 242, 249, 262, 267, 269, 302, 364, 366, 376, 377, 384, 391, 400, 401, 408, 413, 429, 430, 437, 443, 480, 505, 535, 597, 689

POTASH 73, 78, 201, 330, 411, 413, 625, 670, 682

POTASSIUM 45, 165, 168, 377

POTASSIUM COMPOUNDS 45

PRASEODYMIUM 144 550

PRECIPITATION, CHEMICAL 58, 78, 191, 269, 288, 311, 339, 565, 596, 600, 613, 626, 686, 655

PRECIPITATION, METEOROLOGICAL 210, 268, 306, 309, 456, 8[illegible]

PREDICTIONS 93, 98, 99, 100, 129, 197, 222, 233, 248, 273, 277, 279, 297, 308, 362, 405, 477, 505, 506, 522, 523, 532, 538, 579, 590, 690, 691, 694, 701, 702, 704, 705

PRESSURE EFFECTS 16, 36, 41, 46, 48, 49, 57, 72, 78, 79, 86, 89, 91, 103, 108, 173, 360, 423, 447, 468, 480, 536, 577, 641

PROGRAMS, GOVERNMENT 503

PROMETHIUM 147 590

PROTACTINIUM 585

PUMPS 650, 670

PUREX PROCESS 336, 424, 589, 603, 689, 698

PYRITE 65

*** Q ***

QUALITY ASSURANCE 52, 499

QUARTZ 48, 66, 356, 606

QUARTZITES 63, 68

*** R ***

RADIATION DETECTORS 106, 245, 555, 631, 650

RADIATION DOSE 82, 150, 209, 238, 2[illegible], 268, 271, 275, 283, 297, 308, 309, 346, 387, 359, 366, 498, 500, 501, 512, 565, 651

RADIATION EFFECTS 7, 8, 9, 12, 30, 41, 82, 86, 78, 82, 99, 108, 133, 206, 228, 268, 309, 310, 316, 339, 345, 351, 360, 361, 406, 421, 468, 477, 512, 596, 6[illegible], 683, 661

RADIATION HAZARDS 152, 233, 253, 325, 326, 339, 383, 385, 436, 488, 506, 526, 565, 700

RADIATION SOURCES 138, 424, 590

RADIATION, BACKGROUND 498

RADIATION, GAMMA 82, 98, 100, 106, 168, 166, 208, 280, 283, 335, 377, 526, 577, 590, 613, 656, 678, 698

RADIOACTIVE DECAY 17, 80, 78, 123, 125, 238, 258, 318, 468, 498, 573, 650, 698

RADIOACTIVITY 3, 170, 229, 258, 305, 598, 619, 626, 650, 700, 702

RADIOCHEMISTRY 47, 68, 208, 337, 377, 458

RADIOECOLOGY 255, 555

RADIOLYSIS 83, 86, 87, 78, 91, 96, 100, 108, 102, 103, 202, 246, 468, 490, 577

RADIONUCLIDE MIGRATION 8, 9, 11, 18, 15, 29, 31, 32, 32, 37, 39, 65, 73, 137, 157, 162, 163, 178, 179, 181, 186, 192, 196, 198, 200, 203, 215, 217, 218, 219, 220, 221, 222, 225, 226, 227, 229, 230, 231, 232, 233, 234, 236, 237, 280, 281, 283, 287, 248, 249, 250, 251, 252, 258, 255, 257, 259, 262, 268, 265, 268, 270, 272, 276, 279, 281, 282, 284, 285, 287, 289, 290, 291, 292, 297, 298, 300, 301, 302, 303, 304, 305, 306, 307, 309, 311, 312, 314, 315, 319, 328, 330, 333, 335, 336, 344, 347, 356, 369, 376, 383, 402, 415, 444, 485, 548, 552, 582

RADIONUCLIDES 58, 73, 120, 157, 166, 175, 217, 219, 231, 233, 235, 237, 241, 248, 247, 258, 261, 269, 276, 277, 300, 302, 312, 327, 335, 356, 358, 377, 428, 468, 505, 590, 663, 680, 700, 703

RADIUM 188, 303

RADIUM 226 258, 261, 268, 318, 337

RADON 328, 437

RADON GASES 501

RAFFINATES 117, 666, 671

RARE EARTH COMPOUNDS 207, 268

RARE EARTHS 2, 118, 197, 679

REACTOR BLANKET 597, 699

REACTORS 1, 168, 175, 220, 287, 326, 332, 337, 509, 512, 516, 520, 583, 609, 692, 698, 705

REACTORS, BOILING WATER 135, 280, 322, 520, 690, 691

REACTORS, BREEDER 84, 523

REACTORS, CANDU 125, 318, 473, 520

REACTORS, FUSION 697, 699, 701

REACTORS, HIGH-TEMPERATURE GAS-COOLED 125, 238, 280, 500, 501, 502, 520, 585, 698, 695

REACTORS, LIGHT WATER 88, 125, 135, 183, 258, 256, 281, 321, 322, 501, 552, 590, 592, 621, 690, 691, 698, 700

REACTORS, LIGHT WATER BREEDER 251

REACTORS, LIQUID METAL FAST BREEDER 96, 258, 568, 698

REACTORS, NATURAL 15, 65, 231, 236, 548

REACTORS, PRESSURIZED WATER 135, 280, 322, 520, 690, 691

RECEPTIVITY 368

RECHARGE 222, 397

RECOMMENDATIONS 13, 55, 106, 211, 217, 277, 316, 317, 321, 325, 352, 376, 383, 413, 441, 489, 451, 457, 458, 471, 477, 488, 493, 495, 503, 509, 512, 516, 552, 555, 557, 558, 559, 5[illegible], 601

RECONCENTRATION 251, 318, 333

RECRYSTALLIZATION 59

RECYCLING 328, 520, 550, 652, 700

REDOX REACTIONS 11, 269, 601

REGULATIONS 3, 205

REGULATIONS, FEDERAL 127, 181

RELEASE LIMITS 181, 188, 277, 303, [illegible] 8

REMOTE HANDLING 212, 537

REPOSITORY 3, 7, 9, 10, 11, 12, 13, 16, 18, [illegible] 20, 21, 22, 23, 24, 26, [illegible], 41, 42, 48, 51, 52, 54, 55, 5[illegible] 59, 61, 62, 66, 67, 68, 77, 79, 80, 82, 84, 88, 94, 109, 120, 137, 140, 146, 150, 151, 156, 157, 161, 168, 170, 177, 190, 218, 237, 241, 256, 259, 262, 263, 268, 279, 282, 296, 299, 309, 312, 313, 315, 316, 317, 319, 328, 329, 334, 337, 340, 341, 345, 346, 347, 352, 353, 359, 361, 362, 368, 369, 371, 372, 373, 374, 375, 380, 381, 382, 384, 385, 387, 388, 389, 390, 393, 395, 403, 404, 406, 407, 413, 423, 425, 426, 427, 429, 431, 435, 439, 439, 440, 442, 446, 447, 450, 451, 457, 458, 459, 466, 469, 470, 475, 482, 487, 488, 489, 491, 492, 493, 494, 496, 497, 503, 504, 505, 517, 525, 527, 528, 529, 530, 531, 532, 533, 534, 536, 538, 539, 541, 542, 543, 545, 546, 547, 548, 549, 550, 552, 553, 554, 555, 556, 557, 558, 560, 561, 562, 563, 599, 693, 702

REPROCESSING 211, 285, 256, 313, 327, 386, 457, 500, 501, 502, 523, 572, 588, 693, 695

RESIDUES 333, 490, 616

RESINS 83, 84, 136, 178, 339, 409, 486, 600, 616, 621, 623, 636, 654, 670, 687, 688

RESISTIVITY 73, 437, 656

RESUSPENSION 238, 261, 249, 309

RETARDATION FACTOR 256, 257, 262, 269, 276, 302

RETENTION PONDS 175, 197, 320

RETRIEVABILITY 125, 137, 146, 158, 184, 185, 186, 204, 206, 212 225, 308, 309, 313, 329, 335, 380, 381, 382, 385, 386, 387, 372, 376, 392, 473, 475, 482, 515, 525, 526, 528, 529, 585, 550, 551, 552, 557, 559, 561, 588, 630, 671, 683, 693

REVIEWS 15, 19, 21, 22, 23, 24, 38, 41, 68, 71, 143, 150, 176, 217, 218, 219, 230, 231, 236, 250, 261, 268, 310, 316, 321, 322, 323, 326, 327, 334, 337,

KEYWORD INDEX

REVIEWS 338, 348, 351, 353, 355, 356, 362, 378, 379, 380, 384, 385, 402, 408, 412, 413, 425, 431, 432, 438, 441, 449, 453, 459, 483, 484, 489, 491, 492, 493, 498, 503, 505, 511, 514, 519, 524, 536, 564, 582, 586, 610, 611, 613, 624, 663, 684

RECEIVER 106 285, 590

RHIOLITES 219

RIVERS 197, 289, 309, 337, 339, 420, 484

ROCK MECHANICS 18, 19, 20, 21, 22, 23, 24, 51, 55, 56, 69, 80, 85, 86, 90, 140, 174, 196, 218, 237, 312, 313, 316, 351, 361, 373, 374, 380, 383, 384, 394, 398, 411, 413, 414, 418, 425, 435, 440, 447, 458, 466, 469, 485, 489, 507, 508, 545, 552, 558, 632

ROCK-FLUID INTERACTIONS 29, 272, 638

ROCKS 18, 19, 20, 21, 22, 23, 24, 28, 32, 49, 69, 73, 77, 83, 90, 180, 193, 194, 195, 196, 217, 232, 234, 241, 249, 250, 262, 264, 269, 297, 299, 312, 325, 342, 355, 356, 368, 375, 377, 382, 384, 391, 396, 398, 399, 400, 401, 411, 414, 425, 427, 429, 430, 442, 443, 464, 466, 470, 473, 480, 491, 498, 535, 536, 543, 546, 547, 548, 549, 552, 582

ROGERS 283

ROCTS 268, 310

RUFIDION 58, 181, 155, 221

RUHOFF 173, 210, 389, 456

RUTHENIUM 65, 268, 637, 660, 689

RUTHENIUM 106 238, 285, 501, 588, 590, 620

RUTILE 597

*** S ***

SABOTAGE 260, 498, 506, 512

SAFETY 18, 101, 110, 120, 122, 125, 176, 186, 211, 221, 222, 312, 313, 315, 319, 322, 330, 332, 333, 380, 383, 369, 391, 402, 437, 433, 438, 436, 498, 497, 498, 506, 512, 516, 522, 530, 541, 542, 553, 558, 557, 560, 568, 588, 678

SALINITY 357

SALT DEPOSITS 5, 9, 10, 12, 13, 15, 16, 20, 25, 41, 42, 50, 51, 54, 55, 67, 68, 72, 73, 75, 79, 80, 81, 82, 86, 102, 103, 109, 137, 140, 150, 151, 161, 170, 177, 202, 231, 237, 240, 241, 256, 259, 279, 300, 313, 316, 318, 319, 328, 329, 330, 337, 348, 355, 360, 361, 362, 368, 373, 375, 378, 379, 383, 384, 385, 386, 391, 394, 395, 396, 402, 407, 411, 413, 417, 425, 426, 427, 428, 429, 431, 439, 440, 442, 443, 450, 453, 458, 456, 458, 459, 462, 463, 470, 471, 473, 478, 479, 482, 497, 529, 530, 531, 532, 533, 536, 537, 539, 540, 545, 546, 550, 552, 553, 557, 561, 562, 579, 669

SALT DOMES 15, 24, 42, 61, 68, 75, 103, 150, 177, 263, 312, 313, 337, 348, 353, 359, 373, 374, 407, 412, 413, 420, 425, 428, 430, 432, 436, 439, 457, 460, 461, 472, 485, 528, 540, 541, 552

SALTS 9, 31, 39, 59, 73, 77, 82, 129, 170, 193, 198, 216, 237, 273, 278, 325, 336, 347, 425, 468, 486, 522, 574, 575, 586, 596, 603, 612, 615

SAMARIUM 151 590

SAMPLES 35, 61, 63, 166, 167, 168, 173, 181, 196, 204, 227, 307, 468, 692

SAMPLING 45, 158, 210, 290, 335, 594

SANDS 25, 59, 66, 132, 173, 183, 192, 291, 293, 302, 365, 430, 460

SANDSTONES 39, 50, 63, 65, 174, 218, 219, 241, 299, 300, 368, 365, 379, 393, 394, 399, 401, 408, 413, 429, 433, 443

SATURATED ZONE 215, 219, 251, 265, 376

SATURATION 183

SCHEELITE 188

SCHISTS 5, 68, 289, 425, 484

SCINTILLATION COUNTERS 166, 285

SEA DISPOSAL 37, 62, 102, 103, 110, 145, 188, 230, 260, 261, 281, 303, 308, 317, 328, 337, 351, 356, 357, 367, 415, 449, 465, 476, 477, 495, 515, 517, 518, 557, 599, 682

SEAL MATERIALS 136, 210, 536

SEALING 52, 59, 66, 491, 536

SEAWATER 103, 230, 281, 285, 337, 357, 402, 415, 477

SEDIMENTARY ROCKS 11, 16, 17, 22, 28, 29, 31, 48, 50, 52, 57, 63, 65, 66, 68, 75, 87, 89, 189, 195, 213, 214, 218, 219, 241, 249, 269, 300, 353, 368, 365, 375, 378, 379, 384, 385, 390, 391, 394, 395, 400, 401, 402, 408, 412, 413, 421, 423, 425, 427, 429, 432, 439, 442, 443, 453, 480, 483, 486, 489, 579, 681

SEDIMENTATION 311

SEDIMENTS 38, 52, 66, 102, 103, 162, 180, 183, 189, 193, 194, 195, 197, 199, 201, 210, 217, 230, 232, 238, 255, 261, 263, 285, 289, 290, 297, 308, 325, 342, 358, 356, 357, 365, 366, 376, 379, 389, 390, 403, 404, 415, 420, 425, 428, 430, 444, 477, 591

SEEPAGE 277, 486

SEEPAGE FRACTURES 173

SEEPAGE PITS 178, 391, 489

SEISMIC REFLECTION 481, 485

SEISMOLOGY 207, 264, 334, 335, 360, 371, 413, 427, 428, 430, 447, 455, 472, 485

SELENIUM 84

SELENIUM 79 264, 594

SEPARATION PROCESSES 167, 285, 332, 428, 468, 585, 594, 601, 607, 609, 623, 681, 702

SERPENTINE 484

SHAFTS 309, 342, 413, 536, 553

SHALES 11, 14, 22, 24, 28, 38, 39, 48, 50, 51, 52, 54, 5 , 66, 68, 75, 76, 77, 87, 89, 189, 174, 178, 213, 214, 219, 237, 244, 269, 279, 282, 299, 312, 313, 322, 355, 360, 365, 371, 373, 375, 379, 383, 385, 391, 393, 394, 395, 401, 402, 408, 411, 413, 421, 423, 425, 427, 429, 430, 442, 443, 458, 478, 483, 528, 546, 548, 552, 553, 558, 563, 575, 601, 602

SHIELDED CONTAINERS 97, 587, 633

SHIELDING 125, 134, 186, 433, 630

SHOTS 393

SILICA 93, 122, 128, 168, 377, 410, 628, 642, 655

SILICATES 14, 17, 30, 40, 88, 113, 118, 126, 131, 157, 236, 256, 282, 286, 580, 602, 605, 606, 607, 608, 622, 648, 665, 650, 669

SILICON 606

SILTS 173, 183, 302, 305, 483

SILTSTONES 38, 393, 483

SILVER 84

SINKHOLES 478

SINTERING 386, 636, 649

SITE EVALUATION 12, 55, 73, 78, 69, 193, 194, 207, 265, 316, 328, 338, 368, 408, 437, 450, 453, 455, 456, 457, 487, 503, 516, 530, 533

SITE PREPARATION 308, 553

SITE SELECTION 17, 18, 19, 21, 22, 23, 24, 26, 34, 36, 48, 53, 68, 73, 76, 88, 88, 102, 137, 146, 171, 225, 237, 281, 261, 268, 265, 302, 303, 308, 312, 313, 314, 315, 318, 319, 321, 328, 329, 333, 382, 388, 350, 352, 353, 355, 356, 359, 360, 362, 363, 364, 365, 366, 374, 375, 378, 379, 380, 381, 382, 384, 385, 389, 390, 391, 393, 397, 398, 399, 400, 401, 402, 403, 408, 408, 413, 417, 423, 426, 427, 429, 430, 431, 434, 435, 436, 437, 438, 439, 442, 443, 444, 446, 447, 448, 450, 452, 453, 454, 455, 456, 457, 458, 459, 462, 468, 470, 471, 480, 482, 484, 489, 491, 491, 492, 494, 503, 504, 508, 515, 516, 528, 535, 535, 546, 552, 557, 628, 632

SITE SURVEILLANCE 114, 117, 166,

KEYWORD INDEX

SITE SURVEILLANCE 167, 168, 189, 196, 210, 230, 335, 480, 515

SKID 230

SLEEVES 561

SLUDGES 46, 60, 96, 107, 139, 144, 179, 212, 213, 335, 339, 347, 367, 467, 480, 576, 586, 576, 577, 596, 613, 615, 620, 622, 630, 634, 655, 656, 657, 658, 660, 673, 676, 681, 683

SLURRIES 391, 394, 399, 400, 642, 654, 666, 670

SMECTITE 14, 483

SOCIAL ASPECTS 310, 318, 320, 333, 391, 516, 544, 564

SODIUM 27, 42, 81, 88, 96, 185, 188, 206, 243, 294, 377, 657

SODIUM COMPOUNDS 468, 608

SOFTWARE 22 65

SOIL TRANSPORT 4, 227, 255, 264, 265, 301, 320

SOILS 5, 7, 8, 9, 23, 85, 98, 162, 165, 170, 173, 178, 179, 181, 183, 186, 189, 191, 193, 194, 195, 198, 199, 200, 220, 210, 216, 217, 220, 227, 229, 232, 230, 255, 256, 265, 268, 265, 270, 272, 284, 286, 288, 292, 293, 294, 295, 301, 302, 305, 306, 307, 308, 308, 309, 310, 312, 320, 332, 344, 347, 350, 420, 428, 472, 475, 501, 605, 608

SOLAR SYSTEM 433, 434

SOLIDIFICATION 3, 5, 40, 44, 64, 115, 116, 122, 126, 131, 145, 164, 190, 245, 256, 261, 264, 330, 347, 355, 392, 413, 467, 498, 509, 522, 523, 572, 573, 575, 576, 580, 582, 583, 586, 592, 593, 596, 597, 606, 607, 608, 611, 614, 615, 619, 621, 625, 627, 632, 635, 636, 638, 639, 644, 646, 649, 653, 658, 661, 662, 664, 665, 666, 669, 671, 673, 674, 675, 676, 683, 686, 693, 702

SOLUBILITY 6, 16, 41, 44, 132, 155, 204, 265, 310, 362, 373, 659

SOLUTES 230

SOLVENTS 43

SORPTION 1, 14, 27, 28, 32, 33, 38, 39, 44, 58, 71, 77, 78, 83, 163, 199, 216, 226, 232, 238, 241, 244, 25., 260, 264, 268, 268, 269, 289, 293, 294, 295, 297, 300, 302, 311, 312, 358, 356, 369, 377, 413, 447, 449, 468, 572, 623, 638, 646, 649, 655, 660, 675, 676, 677, 686, 688

SORPTION COEFFICIENT 73, 241, 251, 252

SPACE DISPOSAL 433, 434, 444, 445

SPACE TECHNOLOGY 433, 434, 448

SPACE VEHICLES 448

SPECIFIC GRAVITY 213

SPECIFIC HEAT 53

SPECTROMETRY 6, 83, 113, 124, 167, 688

SPECTROMETRY, MASS 130

SPENT FUELS 20, 54, 75, 96, 123, 125, 132, 134, 135, 143, 144, 146, 151, 211, 233, 312, 313, 315, 320, 324, 325, 329, 369, 372, 375, 378, 389, 447, 457, 473, 506, 529, 545, 546, 550, 552, 557, 559, 565, 566, 585, 586, 589, 636, 637, 690, 700, 703

SPHERE 149

SPINEL 80, 656

STABILITY, CHEMICAL 157

STABILIZATION, CHEMICAL 17, 30, 80, 637, 640, 668

STABILIZATION, PHYSICAL 17, 268, 593, 688

STABILIZATION, VEGETATIVE 268

STAINLESS STEELS 101, 103, 133, 143, 346, 34., 697, 699

STANDARDS 3

STANDARDS, FEDERAL 84, 187, 209, 321, 324, 339, 363, 457, 493, 499, 505, 508, 511, 513, 518, 522, 524, 526, 527, 564, 610

STANDARDS, INTERNATIONAL 123, 339, 495, 518, 528, 613

STANDARDS, STATE 344, 528

STATISTICS 26, 71, 110, 153, 181, 223, 233, 253, 284, 354, 414, 512, 560

STEAM 377

STEELS 95, 103, 116, 117, 125, 129, 136, 159, 329, 341, 416, 573, 683

STEMS 310

STORAGE COEFFICIENT 192, 444

STORAGE, ABOVEGROUND 144, 158, 164, 176, 184, 187, 205, 206, 318, 320, 327, 329, 450, 473, 478, 491, 573, 693

STORAGE, AWAY-FROM-REACTOR 705

STORAGE, EXTRATERRESTRIAL 433, 434, 440

STORAGE, FACILITIES 218

STORAGE, GEOLOGIC 11, 12, 13, 14, 15, 17, 18, 19, 21, 22, 23, 24, 25, 26, 35, 40, 41, 42, 48, 50, 52, 55, 59, 66, 67, 68, 75, 78, 79, 81, 85, 88, 89, 94, 97, 102, 103, 108, 109, 121, 123, 125, 127, 128, 132, 137, 140, 146, 149, 150, 151, 156, 157, 160, 176, 177, 189, 202, 206, 213, 218, 219, 220, 224, 225, 226, 233, 236, 237, 241, 249, 251, 258, 256, 259, 260, 264, 269, 279, 282, 296, 299, 300, 304, 310, 315, 316, 318, 319, 321, 327, 328, 329, 332, 334, 336, 337, 338, 350, 352, 355, 359, 361, 364, 368, 369, 371, 372, 378, 375, 378, 379, 380, 381, 382, 383, 384, 385, 386, 387, 388, 391, 395, 397, 398, 400, 401, 403, 404, 406, 407, 408, 417, 422, 423, 425, 426, 427, 430, 435, 436, 437, 438, 439, 440, 442, 444, 446, 447, 450, 451, 452, 453, 454, 457, 458, 459, 464, 466, 469, 473, 473, 478, 482, 483, 484, 487, 488, 489, 491, 492, 497, 498, 501, 504, 505, 517, 519, 523, 528, 529, 530, 531, 532, 533, 536, 538, 539, 540, 541, 542, 543, 544, 545, 546, 547, 548, 549, 550, 551, 552, 553, 554, 555, 557, 558, 559, 562, 563, 579, 624, 632, 640, 693, 702

STORAGE, INTERIM 125, 205, 318, 318, 326, 702, 705

STRAIN 10, 36, 49, 73, 82, 85, 86, 143, 373, 446, 447, 466

STRATIGRAPHY 26, 189, 353, 365, 371, 379, 385, 387, 388, 393, 403, 410, 412, 417, 425, 425, 427, 430, 431, 430, 444, 479

STREAMS 166, 255, 311, 425, 652

STRENGTH, COMPRESSIVE 59, 60, 170, 373, 380, 394, 440, 655, 673

STRENGTH, MECHANICAL 18, 52, 79, 389, 398, 440, 678

STRENGTH, TENSILE 174, 384, 440

STRESS 10, 17, 18, 20, 28, 36, 41, 49, 51, 55, 56, 67, 72, 85, 86, 90, 129, 139, 143, 146, 148, 170, 218, 237, 312, 380, 382, 389, 396, 414, 421, 440, 446, 447, 458, 466, 469, 472, 535, 540, 641

STRONTIUM 30, 54, 58, 64, 155, 157, 165, 180, 214, 229, 260, 280, 292, 302, 311, 306, 387, 398, 391, 424, 447, 467, 576, 582, 615, 619, 638, 645, 649, 659, 664, 683, 687, 688

STRONTIUM 85 28, 298, 311

STRONTIUM 89 144

STRONTIUM 90 144, 169, 204, 227, 230, 238, 347, 460, 486, 500, 501, 590, 620, 624, 639

STRUCTURAL STABILITY 28, 51, 67, 353, 359, 360, 427, 436, 438, 439, 450, 500

SUBSIDENCE 260, 479

SULFATES 16, 38, 74, 160, 377

SULFIDES 236

SULFUR COMPOUNDS 139, 250, 363, 608

SULFUR 35 69

SURFACE CONTAMINATION 94, 197, 296, 506

SURFACE PROPERTIES 28, 133, 315

SURFACE WATERS 256, 261, 264, 289, 307, 308, 377, 391, 311, 413, 420, 438, 470

SURFACTANTS 626

SURVEYS 135

SYENITES 150, 372, 645

KEYWORD INDEX

SYLVITE 74

*** T ***

TAILINGS 327, 511, 610

TALC 484

TANKS 107, 117, 129, 139, 162, 168, 193, 194, 195, 197, 198, 199, 221, 222, 295, 335, 367, 468, 470, 573, 583, 587, 589, 603, 630, 633, 654, 667, 668, 671, 680, 683, 692, 693

TECHNETIUM 11, 32, 65, 227, 228, 237, 250, 290, 369

TECHNETIUM 95M 28

TECHNETIUM 99 8, 236, 258, 259, 268, 269, 292, 594

TECTONICS 34, 334, 353, 371, 406, 412, 413, 417, 425, 427, 438, 439, 444

TEMPERATURE 10, 16, 20, 28, 37, 40, 47, 58, 69, 72, 75, 78, 81, 86, 93, 99, 101, 103, 104, 105, 107, 110, 121, 122, 124, 136, 131, 151, 155, 246, 267, 265, 274, 316, 357, 373, 374, 381, 397, 409, 456, 460, 487, 557, 568, 591, 604, 606, 608, 615, 626, 635, 650, 660, 668, 672, 673, 675, 680, 685, 692

THEORETICAL STUDIES 13, 15, 18, 19, 20, 24, 25, 26, 34, 35, 37, 41, 48, 49, 50, 55, 57, 67, 68, 75, 79, 84, 85, 86, 89, 112, 114, 119, 123, 140, 150, 158, 171, 197, 202, 215, 218, 219, 221, 222, 231, 232, 233, 236, 236, 237, 238, 239, 242, 243, 252, 256, 257, 258, 259, 260, 261, 262, 263, 268, 268, 267, 265, 273, 275, 277, 279, 288, 296, 303, 304, 314, 320, 333, 341, 351, 356, 361, 369, 375, 375, 376, 380, 382, 386, 387, 395, 400, 407, 418, 423, 428, 440, 444, 462, 466, 473, 491, 492, 500, 501, 502, 505, 506, 519, 520, 521, 529, 532, 540, 541, 542, 544, 545, 547, 548, 549, 550, 561, 562, 563, 564, 565, 568, 576, 585, 590, 601, 605, 630, 632, 650, 652, 653, 667, 676, 685, 690, 694, 695, 697, 700, 705

THERMAL CONDUCTIVITY [illegible] 53, 57, 59, 60, 61, 63, 73, [illegible] 491, 523, 580, 575

THERMAL DIFFUSIVITY 53

THERMAL EFFECTS 356

THERMAL EXPANSION 67, 69, 76, 341, 423, 448, 568

THERMAL LOAD 175

THERMAL PROPERTIES 12, 13, 14, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 34, 36, 37, 40, 41, 42, 46, 50, 51, 53, 55, 56, 57, 60, 61, 63, 69, 72, 73, 75, 76, 77, 79, 80, 82, 85, 87, 88, 89, 91, 96, 104, 110, 119, 128, 131, 133, 138, 180, 183, 185, 163, 208, 219, 237, 261, 262, 263, 268, 297, 312, 315, 316, 319, 351, 361, 372, 378, 375, 380, 381, 382, 384, 386, 387, 388, 389, 390, 398, 411, 413, 423, 425, 433, 437, 440, 441, 446, 447, 448, 459, 469, 470, 484, 488, 491, 533, 535, 536, 552, 563, 568, 569, 575, 620, 622, 626, 632, 643, 656, 657, 658, 665, 679

THERMODYNAMICS 231

THOREX PROCESS 336, 589

THORIUM 6, 7, 8, 30, 220, 261, 268, 324, 437, 501, 523, 585, 607, 640, 695

THORIUM 229 258, 268

THORIUM 230 318

THORIUM 232 47, 84

THYROID 238, 500, 501, 512

TILL 173, 189, 302, 376

TIME FACTOR 19, 28, 86, 258, 262, 302, 309, 382, 383, 396, 535, 546

TIN 126 261

TITANIUM 132, 148, 636, 656

TITANIUM COMPOUNDS 112, 597, 658

TOPOGRAPHY 181, 207, 262, 413, 420, 427

TORNADOES 176, 309

TOTAL DISSOLVED SOLIDS 397

TOTAL ION CONCENTRATION 397

TOXICITY 35, 84, 253, 310, 339

TRACERS 29, 32, 69, 230, 249, 252

TRANSFER COEFFICIENTS 310

TRANSMISSIVITY 192, 216

TRANSMUTATION 332, 337

TRANSPIRATION 456

TRANSPORTATION 90, 96, 110, 123, 136, 143, 147, 152, 182, 187, 261, 309, 312, 323, 326, 329, 330, 333, 341, 343, 345, 346, 355, 372, 448, 475, 496, 505, 506, 507, 525, 530, 531, 560, 566, 567, 568, 587, 590, 630, 705

TRANSPORTATION, AIR 92

TRANSPORTATION, BARGE 565

TRANSPORTATION, RAIL 112, 308, 323, 341, 565

TRANSPORTATION, TRUCK 342

TRANSURANICS 7, 8, 9, 91, 97, 159, 208, 228, 233, 255, 310, 332, 498, 570, 577, 582, 588, 600, 607, 630, 700

TRENCHES 98, 166, 167, 173, 186, 189, 206, 209, 216, 290, 307, 335, 357, 624

TRITIUM 100, 123, 127, 130, 166, 168, 172, 198, 221, 222, 238, 280, 298, 302, 303, 323, 327, 332, 355, 377, 391, 397, 398, 476, 500, 501, 548, 588, 607, 672, 682, 697

TUFFS 32, 60, 77, 83, 88, 196, 204, 206, 209, 250, 251, 252, 295, 410, 413, 678

TUNNELS 309, 342

*** U ***

ULTRABASICS 484

UNSATURATED ZONE 197, 215, 251, 265, 280, 292

UPTAKE 85, 235, 254, 255, 261, 271, 281, 285, 307, 310, 311, 344

URANINITE 236

URANIUM 15, 30, 38, 54, 65, 125, 126, 143, 157, 159, 204, 220, 236, 247, 248, 255, 260, 322, 327, 377, 397, 437, 460, 476, 499, 523, 552, 583, 585, 623, 640, 656, 682, 690, 693, 700, 702

URANIUM COMPOUNDS 521, 702

URANIUM HEXAFLUORIDE 702

URANIUM 233 47, 83, 251, 260, 308, 309, 501, 605

URANIUM 234 398

URANIUM 235 47, 84, 136, 144

URANIUM 236 144

URANIUM 237 251

URANIUM 238 47, 94, 398

UREA FORMALDEHYDE 369, 621

*** V ***

VADOSE ZONE 162, 198

VALENCE 11, 73, 88, 217, 290, 310, 623

VANADIUM 599

VAPORIZATION 608

VEGETATION 197, 238, 255, 260, 307, 308, 310, 311, 344

VENTILATION 75, 312, 341, 398, 532, 537, 581

VERMICULITE 14, 64, 100, 577, 655, 675

VISCOELASTICITY 178

VISCOSITY 25, 69, 386, 602, 626, 656, 657, 658

VITRIFICATION 103, 128, 150, 205, 233, 261, 268, 350, 435, 490, 552, 573, 577, 586, 586, 587, 602, 610, 615, 620, 683, 688, 656, 657, 658, 659, 679, 685, 696, 702

VOLATILITY 278, 658, 620

VOLCANOES 268, 279, 309, 410

VOLUME REDUCTION 159, 190, 206,

KEYWORD INDEX

VOLUME REDUCTION 352, 335, 349, 351, 416, 486, 490, 569, 570, 571, 574, 579, 580, 594, 587, 587, 588, 592, 595, 599, 616, 618, 622, 625, 638, 649, 651, 652, 660, 667, 668, 671, 672, 678, 688, 702

*** W ***

WASHING 468

WASTE DISPOSAL 3, 5, 12, 13, 15, 16, 17, 28, 29, 30, 38, 35, 37, 40, 43, 49, 50, 72, 78, 81, 91, 94, 97, 102, 103, 104, 119, 123, 125, 129, 132, 141, 142, 145, 150, 151, 156, 161, 162, 166, 177, 179, 179, 191, 192, 197, 198, 200, 203, 206, 209, 213, 214, 216, 218, 220, 225, 226, 231, 235, 237, 247, 251, 258, 256, 259, 261, 263, 268, 265, 283, 284, 290, 291, 300, 307, 309, 313, 314, 315, 316, 317, 318, 319, 321, 322, 323, 324, 325, 327, 328, 329, 331, 338, 335, 336, 337, 338, 342, 345, 347, 351, 352, 355, 357, 363, 364, 365, 366, 367, 372, 375, 376, 379, 382, 362, 387, 388, 389, 390, 391, 392, 393, 394, 395, 399, 400, 401, 402, 404, 407, 408, 413, 421, 426, 427, 429, 435, 437, 441, 442, 443, 445, 449, 450, 451, 452, 455, 460, 463, 464, 465, 470, 471, 472, 473, 480, 482, 484, 491, 492, 494, 495, 498, 505, 507, 509, 511, 513, 514, 516, 518, 519, 522, 524, 525, 526, 528, 529, 530, 533, 535, 540, 543, 544, 546, 547, 548, 549, 550, 551, 552, 556, 559, 561, 562, 563, 566, 580, 605, 621, 624, 627, 632, 638, 640, 641, 691, 692, 698

WASTE FORM 663, 685

WASTE MANAGEMENT 12, 13, 59, 94, 97, 109, 112, 123, 143, 146, 149, 150, 152, 158, 161, 164, 165, 171, 176, 179, 180, 185, 187, 189, 190, 205, 206, 210, 212, 218, 220, 225, 226, 232, 238, 236, 241, 247, 251, 259, 278, 300, 307, 308, 309, 313, 316, 317, 320, 321, 322, 323, 324, 325, 326, 327, 328, 329, 330, 331, 333, 334, 335, 338, 339, 340, 341, 343, 344, 346, 347, 352, 369, 372, 376, 403, 415, 419, 442, 452, 457, 458, 459, 467, 471, 473, 474, 475, 476, 477, 490, 493, 495, 503, 505, 508, 509, 510, 511, 514, 515, 516, 519, 522, 523, 524, 525, 527, 529, 533, 538, 544, 550, 551, 553, 558, 555, 556, 557, 558, 559, 561, 566, 579, 580, 582, 583, 586, 587, 588, 589, 590, 592, 593, 603, 604, 608, 610, 613, 615, 618, 627, 636, 645, 650, 653, 658, 663, 667, 668, 669, 670, 671, 672, 676, 682, 683, 688, 690, 691, 693, 694, 695, 696, 697, 702, 703, 704, 705

WASTE PROCESSING 17, 40, 122, 206, 296, 309, 336, 341, 342, 343, 385, 419, 468, 475, 525, 580, 582, 603, 621, 627, 641, 658, 661, 664, 685, 702

WASTE REPRIVAL 517

WASTE STORAGE 3, 5, 15, 16, 17, 24, 26, 30, 40, 41, 42, 44, 47, 50, 72, 75, 91, 93, 94, 97, 98, 99, 100, 101, 102, 103, 107, 108, 116, 119, 125, 127, 128, 132, 135, 139, 140, 141, 144, 146, 150, 151, 154, 156, 158, 159, 162, 168, 177, 184, 185, 187, 197, 198, 199, 203, 204, 206, 208, 209, 210, 212, 214, 224, 225, 226, 236, 237, 241, 249, 251, 258, 235, 300, 305, 312, 318, 315, 316, 317, 318, 320, 321, 323, 324, 325, 326, 327, 329, 334, 335, 336, 337, 338, 343, 341, 345, 346, 347, 347, 351, 352, 363, 364, 365, 368, 375, 378, 379, 383, 386, 387, 388, 389, 390, 391, 392, 394, 397, 399, 400, 401, 405, 407, 408, 413, 416, 418, 426, 427, 429, 435, 436, 437, 438, 439, 442, 443, 444, 450, 451, 457, 458, 459, 464, 468, 470, 477, 480, 482, 484, 487, 488, 491, 492, 493, 494, 496, 499, 503, 508, 513, 516, 519, 520, 523, 524, 529, 532, 537, 538, 540, 543, 544, 546, 547, 548, 549, 550, 552, 553, 554, 555, 557, 558, 559, 561, 562, 563, 566, 573, 579, 580, 581, 582, 583, 587, 589, 590, 593, 599, 601, 610, 621, 624, 627, 630, 631, 643, 645, 661, 667, 668, 669, 671, 681, 683, 687, 692, 693, 700, 703, 705

WASTE TREATMENT 68, 122, 139, 173, 206, 236, 238, 256, 318, 321, 323, 324, 325, 326, 331, 333, 336, 337, 338, 346, 349, 391, 405, 413, 419, 420, 452, 463, 468, 476, 486, 500, 501, 502, 508, 520, 521, 523, 529, 544, 553, 569, 570, 572, 578, 580, 583, 588, 592, 593, 594, 601, 604, 605, 607, 609, 611, 612, 613, 615, 617, 619, 622, 623, 624, 625, 626, 629, 634, 636, 642, 644, 646, 647, 649, 652, 653, 654, 662, 663, 665, 666, 668, 669, 672, 674, 675, 678, 682, 684, 688, 689, 692, 693, 694, 696, 701

WASTE VOLUME 46, 47, 120, 179, 186, 192, 200, 201, 209, 213, 267, 268, 380, 381, 382, 387, 351, 470, 475, 476, 480, 537, 541, 566, 580, 587, 599, 615, 618, 630, 639, 642, 690, 691, 692, 693, 695, 696, 697, 702, 705

WASTE WATER RESERVOIRS 197

WASTE-ROCK INTERACTIONS 10, 11, 32, 36, 39, 40, 50, 57, 58, 71, 72, 78, 79, 80, 81, 82, 85, 87, 88, 89, 137, 189, 157, 203, 217, 228, 236, 240, 249, 251, 312, 315, 334, 375, 383, 406, 421, 423, 427, 442, 446, 468, 472, 480, 499, 508

WASTES, COMMERCIAL 102, 255, 267, 326, 367, 398, 399, 400, 429, 468, 480, 521, 551, 621, 690, 695

WASTES, DEFENSE 102, 206, 427

WASTES, GASEOUS 181, 280, 391, 524, 583, 610, 633, 641, 691

WASTES, HIGH-LEVEL 1, 11, 15, 16, 17, 19, 20, 24, 26, 30, 35, 36, 37, 40, 41, 42, 50, 56, 60, 69, 75, 77, 81, 83, 87, 88, 89, 101, 102, 107, 108, 112, 114, 116, 117, 119, 12, 121, 123, 124, 127, 128, 129, 137, 139, 145, 146, 150, 154, 155, 157, 158, 159, 161, 164, 169, 173, 188, 190, 197, 199, 203, 205, 211, 212, 221, 222, 233, 237, 246, 249, 254, 256, 259, 260, 261, 263, 268, 293, 300, 303, 304, 315, 319, 319, 321, 322, 323, 324, 327, 329, 330, 333, 346, 347, 352, 355, 356, 357, 358, 359, 360, 362, 369, 375, 379, 386, 391, 392, 402, 403, 404, 405, 407, 413, 416, 424, 435, 444, 445, 447, 450, 467, 468, 471, 474, 477, 482, 484, 491, 496, 497, 498, 501, 505, 506, 507, 508, 509, 511, 513, 514, 515, 518, 520, 522, 523, 524, 526, 529, 537, 538, 545, 546, 550, 551, 552, 557, 561, 569, 573, 575, 579, 580, 582, 583, 585, 586, 587, 589, 590, 591, 598, 599, 600, 601, 602, 603, 604, 605, 607, 608, 609, 610, 611, 613, 614, 615, 617, 619, 620, 624, 625, 627, 628, 630, 632, 633, 635, 636, 643, 644, 645, 646, 649, 656, 659, 660, 661, 666, 667, 668, 669, 670, 671, 673, 674, 680, 681, 684, 695, 686, 687, 688, 689, 692, 693, 695, 700, 702, 703

WASTES, INTERMEDIATE-LEVEL 120, 121, 127, 146, 161, 168, 177, 179, 187, 197, 203, 213, 214, 256, 323, 391, 392, 404, 407, 450, 496, 507, 509, 522, 523, 524, 525, 545, 550, 552, 557, 561, 574, 583, 589, 596, 598, 612, 625, 638, 641, 646, 660, 664, 666, 681, 693, 702

WASTES, LIQUID 17, 34, 43, 78, 100, 116, 117, 129, 133, 162, 164, 165, 178, 179, 191, 192, 197, 196, 199, 200, 203, 211, 218, 222, 267, 280, 281, 283, 290, 292, 295, 296, 326, 332, 338, 347, 351, 358, 359, 360, 364, 365, 366, 367, 391, 392, 394, 395, 400, 401, 405, 408, 416, 419, 429, 443, 464, 467, 471, 476, 480, 486, 490, 526, 501, 522, 523, 528, 572, 573, 577, 579, 580, 581, 582, 583, 586, 587, 589, 592, 593, 595, 596, 598, 599, 603, 605, 607, 610, 615, 617, 628, 625, 630, 632, 633, 635, 636, 638, 639, 641, 644, 647, 649, 652, 653, 658, 660, 668, 666, 668, 669, 671, 672, 679, 680, 681, 682, 686, 688, 689, 691, 692, 695, 697, 698, 702

WASTES, LOW-LEVEL 120, 142, 146, 147, 161, 168, 166, 169, 173, 177, 180, 197, 199, 200, 203, 205, 224, 256, 277, 283, 302, 315, 323, 326, 339, 391, 402, 404, 407, 450, 471, 475, 501, 507, 509, 511, 514, 515, 522, 524, 526, 529, 545, 550, 552, 550, 557, 566, 584, 589, 599, 610, 612, 613, 622, 624, 625, 633, 646, 666, 667, 668, 672, 680, 681, 687, 691, 692, 693, 695, 702

WASTES, NONRADIOACTIVE 279, 296, 308, 367, 465, 652

WASTES, ORGANIC 7, 8, 91, 167, 296, 367, 596, 658, 695

KEYWORD INDEX

WASTES, RADIOACTIVE 1, 3, 5, 7, 8, 12, 13, 14, 16, 17, 18, 19, 20, 23, 83, 82, 83, 84, 85, 54, 58, 60, 72, 78, 81, 82, 84, 85, 88, 89, 91, 94, 95, 101, 102, 108, 108, 109, 111, 122, 129, 132, 138, 139, 141, 142, 143, 146, 149, 149, 151, 152, 155, 156, 157, 159, 162, 165, 171, 176, 177, 178, 182, 184, 190, 191, 192, 197, 199, 195, 201, 202, 206, 210, 220, 221, 222, 225, 226, 231, 232, 234, 237, 244, 247, 249, 250, 255, 258, 259, 269, 272, 278, 279, 283, 281, 282, 286, 292, 295, 300, 303, 304, 312, 313, 314, 316, 317, 318, 319, 321, 323, 326, 328, 330, 331, 333, 336, 337, 338, 339, 352, 355, 357, 361, 362, 364, 366, 367, 368, 369, 375, 378, 379, 387, 388, 391, 392, 393, 398, 399, 400, 402, 403, 406, 407, 413, 415, 417, 416, 421, 422, 425, 427, 429, 433, 434, 436, 441, 442, 445, 446, 447, 448, 449, 450, 452, 457, 458, 459, 463, 464, 465, 466, 470, 473, 476, 486, 487, 488, 490, 493, 494, 495, 497, 503, 504, 506, 507, 509, 510, 511, 513, 514, 515, 516, 521, 522, 524, 526, 527, 528, 533, 536, 542, 543, 546, 547, 548, 549, 550, 551, 552, 553, 554, 555, 556, 557, 558, 559, 562, 563, 564, 566, 567, 570, 571, 572, 575, 577, 578, 579, [illegible], 582, 588, 589, 592, 593, 603, 608, 610, 613, 617, 618, 621, 624, 626, 627, 630, 640, 642, 644, 646, 651, 652, 653, 654, 655, 656, 657, 661, 663, 664, 665, 668, 669, 670, 671, 675, 676, 679, 681, 682, 683, 684, 687, 688, 689, 690, 691, 696, 697, 699, 701, 704, 705

WASTES, SOLID 3, 12, 13, 36, 83, 95, 97, 101, 109, 112, 115, 119, 121, 123, 127, 129, 130, 133, 136, 137, 141, 147, 148, 187, 196, 199, 202, 205, 209, 211, 212, 246, 258, 260, 263, 268, 291, 296, 313, 316, 328, 333, 339, 349, 351, 352, 355, 356, 360, 361, 364, 367, 391, 393, 398, 399, 402, 405, 406, 413, 416, 419, 429, 483, 476, 482, 490, 522, 523, 524, 539, 531, 533, 539, 541, 552, 570, 573, 577, 581, 582, 583, 584, 586, 588, 593, 595, 598, 599, 607, 608, 610, 613, 616, 624, 625, 630, 633, 636, 640, 647, 650, 651, 663, 669, 672, 691, 682, 693, 695, 697, 702

WASTES, TRANSURANIC 7, 8, 9, 46, 47, 51, 52, 97, 102, 121, 127, 133, 136, 147, 159, 160, 164, 181, 184, 185, 186, 187, 190, 197, 199, 206, 207, 208, 209, 210, 228, 268, 291, 294, 307, 308, 309, 315, 321, 327, 340, 381, 382, 383, 384, 385, 386, 356, 419, 475, 482, 494, 505, 507, 514, 515, 522, 524, 525, 526, 537, 545, 550, 552, 570, 571, 588, 559, 610, 622, 638, 650, 693, 702

WATER 48, 57, 87, 91, 100, 125, 164, 153, 166, 167, 169, 181, 193, 196, 201, 259, 301, 305, 312, 329, 461, 463, 489, 600, 620, 631, 659, 660, 697

WATER TABLE 24, 162, 190, 193, 194, 195, 197, 199, 221, 222, 202, 485, 459, 473, 645

WAVES 299

WEATHERING 30, 173, 293, 536

WELDSITE 282, 287

WELLS 123, 162, 166, 170, 191, 198, 200, 222, 282, 309, 365, 396, 393, 397, 412, 418, 428, 459, 478, 485

WELLS, INJECTION 5, 192, 213, 214, 267, 355, 365, 391, 399, 400, 401, 408, 499, 536, 632, 681

WISDS 196, 238, 269, 308

WIPP 7, 8, 9, 46, 47, 72, 73, 78, 96, 109, 137, 160, 223, 228, 281, 279, 300, 330, 363, 383, 404, 431, 446, 453, 453, 454, 455, 456, 482, 494, 528, 532, 537

WITHDRAWAL 222

WOOD 136

*** X ***

XENON 355

XENON 135 628

*** Y ***

YTTERBIUM 229

YTTRIALITE 30

YTTRIUM 90 590, 628

*** Z ***

ZEOLITES 2, 77, 181, 286, 410, 467, 486, 572, 606, 619, 649, 668, 672, 687

ZINC 416

ZINC 65 588

ZIRCALOYS 97, 416, 588, 628, 672

ZIRCONIA 185

ZIRCONITE 116, 117, 188, 158, 204, 602, 689

ZIRCONIUM 93 264

ZIRCONIUM 95 285

SUBJECT CATEGORY INDEX

*** C ***

Chemical and Physical Aspects 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, 71, 72, 73, 74, 75, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 102, 155, 156, 160, 165, 174, 202, 217, 218, 231, 241, 247, 262, 263, 282, 291, 293, 298, 300, 366, 378, 392, 387, 398, 397, 423, 440, 442, 447, 448, 481, 533, 563, 637, 638, 641, 675, 676, 677

Container Design and Performance 83, 91, 92, 93, 94, 95, 96, 97, 98, 99, 100, 101, 102, 103, 104, 105, 106, 107, 108, 109, 110, 111, 112, 113, 114, 115, 116, 117, 118, 119, 120, 121, 122, 123, 124, 125, 126, 127, 128, 129, 130, 131, 132, 133, 134, 135, 136, 137, 138, 139, 140, 141, 142, 143, 144, 145, 146, 147, 148, 149, 150, 151, 152, 153, 154, 157, 158, 159, 160, 196, 205, 421, 506, 548, 625, 635, 644, 645

*** D ***

Disposal Site 97, 144, 161, 162, 163, 164, 165, 166, 167, 168, 169, 170, 171, 172, 173, 174, 175, 176, 177, 178, 179, 180, 181, 182, 183, 184, 185, 186, 187, 188, 189, 191, 192, 193, 194, 195, 196, 197, 198, 199, 200, 201, 202, 203, 204, 205, 206, 207, 208, 209, 210, 211, 212, 213, 214, 216, 295, 305, 308, 309, 419, 522, 525, 598, 680, 693

*** E ***

Environmental Transport 4, 14, 15, 73, 80, 162, 163, 166, 167, 168, 179, 181, 188, 192, 197, 198, 200, 201, 215, 216, 217, 218, 219, 220, 221, 222, 223, 224, 225, 226, 227, 228, 229, 230, 231, 232, 233, 234, 235, 236, 237, 238, 239, 240, 241, 242, 243, 244, 245, 246, 247, 248, 249, 250, 251, 252, 253, 254, 255, 256, 257, 258, 259, 260, 261, 262, 263, 264, 265, 266, 267, 268, 269, 270, 271, 272, 273, 274, 275, 276, 277, 278, 279, 280, 281, 282, 283, 284, 285, 286, 287, 288, 289, 290, 291, 292, 293, 294, 295, 296, 297, 298, 299, 300, 301, 302, 303, 304, 305, 306, 307, 308, 309, 310, 311, 318, 362, 369, 376, 383, 415, 490, 500

*** G ***

General Studies 90, 190, 312, 313, 314, 315, 316, 317, 318, 319, 320, 321, 322, 323, 324, 325, 326, 327, 328, 329, 330, 331, 332, 333, 334, 335, 336, 337, 338, 339, 340, 341, 342, 343, 344, 345, 346, 347

Geology, Hydrology and Site Resources 7, 8, 9, 12, 13, 19, 21, 22, 23, 26, 34, 48, 65, 71, 137, 150, 172, 178, 180, 196, 218, 237, 240, 304, 338, 348, 350, 351, 352, 353, 354, 355, 356, 357, 359, 360, 361, 362, 363, 364, 365, 366, 367, 368, 369, 370, 371, 372, 373, 374, 375, 376, 377, 378, 379, 380, 381, 382, 383, 384, 385, 386, 387, 389, 389, 390, 391, 393, 394, 395, 396, 397, 398, 399, 400, 401, 402, 403, 404, 406, 407, 408, 410, 411, 412, 413, 414, 415, 417, 418, 420, 421, 422, 423, 425, 426, 427, 428, 429, 430, 431, 432, 433, 434, 435, 436, 437, 438, 439, 440, 441, 442, 443, 444, 445, 446, 447, 448, 449, 450, 451, 452, 453, 454, 455, 456, 457, 458, 459, 460, 461, 462, 463, 464, 465, 466, 469, 470, 471, 472, 473, 477, 478, 479, 480, 481, 482, 483, 484, 485, 487, 488, 489, 491, 492, 494, 499, 538, 536, 553, 557, 558, 559, 579

*** R ***

Regulatory and Economic Aspects 3, 152, 169, 211, 277, 321, 328, 325, 396, 493, 495, 496, 497, 498, 499, 500, 501, 502, 503, 504, 505, 506, 507, 508, 509, 510, 511, 512, 513, 514, 515, 516, 517, 518, 519, 520, 521, 522, 523, 524, 525, 526, 527, 528, 610, 685

Repository Design and Engineering 20, 42, 52, 59, 66, 75, 123, 140, 156, 157, 177, 225, 226, 255, 328, 361, 372, 386, 387, 388, 389, 395, 457, 466, 491, 490, 529, 530, 531, 532, 533, 534, 535, 536, 537, 538, 539, 540, 541, 542, 543, 544, 545, 546, 547, 548, 549, 550, 551, 552, 553, 554, 555, 556, 557, 558, 559, 560, 561, 562, 563, 632

*** T ***

Transportation Technology 507, 564, 565, 566, 567, 569

*** W ***

Waste Production 172, 601, 610, 690, 691, 692, 693, 694, 695, 696, 697, 699, 699, 700, 701, 702, 703, 704, 705

Waste Treatment 17, 30, 35, 40, 98, 99, 100, 108, 121, 106, 109, 213, 309, 350, 392, 405, 409, 416, 419, 420, 467, 468, 470, 475, 476, 486, 490, 498, 521, 569, 570, 571, 572, 573, 574, 575, 576, 577, 578, 579, 580, 581, 582, 583, 584, 585, 586, 587, 588, 589, 590, 591, 592, 593, 594, 595, 596, 597, 598, 599, 600, 601, 602, 603, 604, 605, 606, 607, 608, 609, 611, 612, 613, 614, 615, 616, 617, 618, 619, 620, 621, 622, 623, 624, 625, 626, 627, 628, 629, 630, 631, 632, 633, 634, 635, 636, 637, 638, 639, 640, 641, 642, 643, 644, 645, 646, 647, 648, 649, 650, 651, 652, 653, 654, 655, 656, 657, 658, 659, 660, 661, 662, 663, 664, 665, 666, 667, 668, 669, 670, 671, 672, 673, 674, 675, 676, 677, 678, 679, 680, 681, 682, 683, 684, 685, 686, 687, 688, 693, 695, 696, 702

PERMUTED TITLE INDEX

Waste Disposal in Salt Deposits - A Bibliography with 000436
Ocean Waste Disposal (A Bibliography with 000367
Retrievable Geologic Storage* *Maximum 000121
Preliminary Concept of Waste Retrieval System-Tank Wall 000630
Management* *Methodology for Estimating 000270
*Operational Techniques to Minimize, Segregate, and 000610
Repository* *An Overview of Material Control and 000120
Repository* *Material Control and 000496
and Plants* * 000050
Nuclear Waste in Natural *Geochemical Constraints on 000015
*Radioactive Wastes in the Federal Republic of Germany: 000596
portesa)* *Influences of Marine Sediment on the 000285
Waste Terminal Storage Program*As Low as is Reasonably 000493
*Liquid Waste Evaporation and 000666
Experiments at Strips* *Data 000056
*Waste Management of 000640
Natural *Geochemical Constraints on Accumulation of 000015
*Synthesis of Nuclear Waste Monazites, Ideal 000640
Waste* *Effect of 000585
Disposal Facilities at Hanford* *Characterization of 000290
*Some Aspects and Long-Term Problems of High-Level and 000035
*Leaching Characteristics of 000153
Test Facilities* *Survey of 000378
Reprocessing Plant* *Management of Highly 000097
as the First Barrier for Long-Term Storage of Highly 000120
*Long Term Geochemical Interactions of High 000157
*Above-Ground Storage Facilities for Vitrified Highly 000573
*Borehole Drilling and Related 000422
*Pacific Northwest Laboratory Monthly 000216
Problem* *Evaluation of the 000699
Salt Vault: A Demonstration of the Disposal of High 000316
*Description of the Energy Research and Development 000325
Waste Isolation* *Role of 000662
*Development of an Effective Aluminosilicate 000606
*Electron Transfer Method of Controlling Th Leaving 000078
Component System* *Preliminary Investigations of the 000165
* 000576
*Development of 000225
90 Migration* *Radionuclides Migration in 000284
*Autonomous Aspiration Unit for Sampling Radioactive 000182
Salt Migration to Dissolution in Salt Domes and How It 000263
*A Critical Study on the International Atomic Energy 000188
Management*Description of the Environmental Protection 000324
of Liquid Radioactive Wastes Using Mechanically 000592
*Critical Pathways of Radionuclides to Man from 000255
Defense Transuranic Waste at the Savannah River Plant, 000206
of Prototype High Level Waste Canister SS-9 While in 000101
*Forced- 000532
Plastic Analysis of Waste-Container Sleeve: 4. 000140
of Criteria for Packaging Plutonium for Transport by 000092
Aspiration Unit for Sampling Radioactive Aerosols from 000182
Sale, Porters Creek Clay, and Tazoo Clay in Parts of 000830
Management Programs and Capabilities within the Los 000171
Capabilities for Alpha Bearing Wastes at the Los 000619
Storage Program* *As Low as is Reasonably Achievable (000493
by the Cassitis Nuclear Explosion, Amchitka Island, 000377
Sediment on the Accumulation of Radionuclides by Green 000285
*Ion Exchange Decontamination of 000486
Storage Concepts* *Description of 000205
Disposal Concepts Regarding Realistic Modeling that 000517
Laboratory* *Waste Management Capabilities for 000619
Savannah River Plant Waste* *Effect of Internal 000099
Storage Facilities* *Design Aspects of the 000581
Site Selection for Repositories of Solid High-Level and 000813
Techniques to Minimize, Segregate, and Account for 000378
Glasses* *Helios Formation from 000131
Radioactive Waste Disposal* *Pollucite and Its 000105
Disposal* *Natural Mineral 000030
*Deep Geologic 000557
for the Disposal in Bedded Salt of LWR Wastes from 000050
* 000318
Transuranic Waste Management* *Identification of 000675
Transuranic Waste at the Savannah River Plant, Aiken, * 000206
Level Radioactive Waste Idaho Chemical Processing * 000366
Level Radioactive Waste Hanford Reservations, * 000387
Level Radioactive Waste* *Summary - 000678
National *Environmental and Other Evaluations of 000380
Nuclear Wastes* *The 000519
*Environmental Effects of 000308
*Clay-Phosphate Ceramics and Vitrometa: 000689
* 000210
Shipment to the Federal Repository* * 000381
Disposal on the Idaho National Engineering Laboratory* * 000382
*Comparisons Among 000385
* *Development of an Effective 000606
*Raman Spectra of Cesium 000113
*Investigations of the Volatility of Cesium from 000138
*Vitrification of High Sodium- 000108
Abstracts* *Radioactive
Abstracts)*
Acceptable Temperatures of Wastes and Containers During
Access Using Underground Mining Techniques for Removal
Accidental Radioactive Releases in Nuclear Waste
Account for Alpha-Bearing Wastes*
Accountability Considerations for a Waste Isolation
Accountability Procedures for a Waste Isolation
Accumulation and Movement of Fission Products in Soils
Accumulation of Actinide Critical Masses from Stored
Accumulation of Radioactive Wastes in the Federal
Accumulation of Radionuclides by Green Algae (Ulva
Achievable (ALARA) Studies Relative to the National
Acid Recovery*
Acquisition, Handling, and Display for the Master
Actinide Contaminated Soil*
Actinide Critical Masses from Stored Nuclear Waste in
Actinide Waste for Geologic Disposal*
Actinide Removal on the Long-Term Hazard of High-Level
Actinide-Bearing Sediments Underlying Liquid Waste
Actinide-Contaminated Spent Fuel Reprocessing Wastes
Actinides from Simulated Reactor Waste Glass*
Active and Inactive Mines for Possible Use as In Situ
Active Decladding Zircaloy Solid Wastes from a Fuel
Active Radioactive Wastes* *Glass
Active Waste with Crystalline Rock Repository Media*
Active Wastes*
Activities at the Stripa Mine*
Activities Report January 1976*
Activity Levels in Fusion Reactor Blankets. Waste
Activity Solidified Wastes in Underground Salt Mines*
Administration Technological Program for Radioactive
Admixtures in Preparing Bonse Cements for Radioactive
Adsorbent for Radioactive Waste Disposal*
Adsorption of Actinides Elements by Geologic Media*
Adsorption of Radiostrontium in Sacley Soil. Three-
Adsorption Studies by the U.S. Geological Survey*
Advanced Waste Management Plans*
Aerated Zones. 3. Theoretical Estimation of Strontium
Aerosols from Air*
Affects a Radioactive Waste Isolation Facility Design*
Agency (IAEA) Definition of High-Level Radioactive
Agency Program for Reprocessing and High Level Waste
Agitated Thin-Film Evaporators* *Volume Reduction
Agro-Ecosystems*
Aiken, South Carolina* for Long-Term Management of
Air and Water Storage* *History
Air Cooling of a Waste Isolation Pilot Plant Mine Drift*
Air Gap Influence on Hole Closure* *Thermoelastic/
Air* *Review
Air* *Autonomous
Alabama, Mississippi and Tennessee as Potential Host
Alamos Scientific Laboratory* *Waste
Alamos Scientific Laboratory* *Waste Management
ALARA) Studies Relative to the National Waste Terminal
Alaska* Distribution in a Cavity and Chimney Produced
Algae (Ulva portesa)* *Influences of Marine
Alkaline Solutions of Nuclear Waste*
Allied-General Nuclear Services On-Site Solid Waste
Allows Objective Risk Assessment to be Made* and Ocean
Alpha Bearing Wastes at the Los Alamos Scientific
Alpha Radiation on Borosilicate Glass Containing
Alpha Repository: 2. Conceptual Layouts of Underground
Alpha-Bearing Wastes in Geological Formations* *
Alpha-Bearing Wastes* *Operational
Alpha-Decay and its Significance for Radioactive Waste
Alteration on Geological Occurrences and in Deep-Burial
Alteration: Its Implication for Radioactive Waste
Alternative Actions*
Alternative Fuel Cycles* *Estimates of Relative Areas
Alternative Waste Management Systems*
Alternatives for Idaho National Engineering Laboratory
Alternatives for Long-Term Management of Defense
Alternatives for Long-Term Management of Defense High-
Alternatives for Long-Term Management of Defense High-
Alternatives for Long-Term Management of Defense High-
Alternatives for Long-Term Management of Stored Idaho
Alternatives for Managing Post-Light-Water Reactor
Alternatives for Managing Transuranic Waste*
Alternatives to Monolithic High Level Waste Products*
Alternatives Involving Leave-in-Place Management*
Alternatives Involving Retrieval, Processing, and
Alternatives Involving Retrieval, Processing, and
Alternatives, Concepts and Models*
Aluminosilicate Adsorbent for Radioactive Waste Disposal
Aluminosilicate Waste Forms*
Aluminosilicates by Mass Spectrometry*
Aluminum Wastes: Composition Ranges and Properties*

and Chimney Produced by the Camphin Nuclear Explosion,
in Geologic Formations in the United States of
Storage in Mined Caverns in Crystalline Rock, @Swedish-
@Retention of Plutonium and
in Southeastern Idaho@ @Plutonium and
@Plutonium-
@Sorption of Technetium 99, Neptunium 237, and
Sorption on Two United States Subsoils from Differing @
Burial, Curium, Europium, Uranium, Plutonium, and
Waste@ @Final Purification and Concentration of
Ratios and Diffusion Coefficients of Neptunium,
@Natural Glasses:
Sludges@ @Sampling and
Isolation in Geologic Formations. Volume 19 - Thermal
Geologic Formations. Volume 22 - Environmental Effluent
Geologic Formations. Volume 2C. Thermomechanical Stress
of the Proposed Salt Mine Repository@ @
Salt Cavities Formed by Solution Mining: 1. Method of
@Radioactive Waste Transportation Systems
@Computer Enhanced "Release Scenario"
Progress Report for Period April 1 to @Waste Management
Utilization of the HOL Data Structure in Consequence
Emphasis on High-Level Waste Management. Volume 2@ @An
Wastes in @A Preliminary Engineering and Economic
Radioactive Waste Repository in a Bedded Salt @An
in the Earth's Crust@ @An
@
Potential Nuclear Waste Repository, Eddy @Review and
Environmental Geochemistry@ @
Method of Analysis and Preliminary Results @Structural
Waste Management@ @Technology Assessment and Impact
Strain Data Suitable for Finite-Element Elastic-Plastic
New York Nuclear Service Center, West @Geotechnical
Fissure Model@ @
Estimates of Loading on the @Thermoelastic/Plastic
Salt Strength on Sleeve Loading@ @Thermoelastic/Plastic
Influence on Hole Closure@ @Thermoelastic/Plastic
Geologic Isolation of Nuclear Waste@ @Program Scenario
@Systems
@Results and
@Hazards
of Vitrified High-Level Reprocessing Waste. 4. Safety
Suitability of Anhydrite as a @P-T-X Relations of
and Brine and Their Implications for the Suitability of
Behavior of Transuranic Elements in Soils, Plants and
@Decommissioning of Nuclear Facilities - An
in Ground Water at the Nevada Test Site - Part 2:
Evaluation of Waste Transport in Selected Rocks: 1977
Waste Glasses@ @
@
Mathematical Simulation of Sediment and Contaminant @
@Basalt Waste Isolation Program
@Studies of Nuclear-Waste Migration in Geologic Media:
@Annual Report on DOWT PROJECT
@Are Plutons the
High-Level Radioactive Wastes@ @
for a International Radionuclide Repository in
High-Level Radioactive Waste@ @
@Radioactive Waste Disposal in
of a Potential Waste Emplacement Site, Salt Valley
Industrial Wastes in Subsurface Rocks on North Flank of
Disposal of Radioactive Wastes @Geologic Summary of the
Feasibility of Talc and Serpentinite Bodies from the
Studies@ @
Repositories of Radioactive Wastes@ @An
Repositories of Radioactive Wastes@ @An
Argillaceous and Crystalline Rocks: Some @An
and Transportation Quarterly Progress Report,
Fuel Cycle-Parts 1 and 2: Progress Report for Period
Laboratories Waste Isolation Pilot Project (WIPP),
@Basalt Waste Isolation Program Monthly Report,
Incineration and Immobilization for Nuclear Facilities (
of Neptunium, Americium, and Curium in Soil-
@Fuming Evaporators to Work: Concentration of
@The Sandia Solidification Process - A Broad Range
@Long-Term Planning for Management of
Storage Potential in Selected Segments of the Mesozoic
Vertical Boreholes and Brine Invasion into Fresh Water
Vertical Boreholes and Brine Invasion into Fresh Water
Vertical Boreholes and Brine Invasion into Fresh Water
Related to the Waste Storage Potential of Mesozoic
Effects of Liquid Waste Disposal in Deep Saline
Numerical Simulation of Brine Invasion of Fresh Water
@Antarctica - A Potential International Burial
and Americium Contamination Near a Transuranic Storage
of Cores from Holes P-3 NM-1 through P-3 NM-5,
in a Desert Pavement-Desert Round Soil System in
@Measuring the Permeability of Slowly Argillite from
Amchitka Island, Alaska@ Distribution in a Cavity
America@ @Waste Isolation
American Cooperative Program on Radioactive Waste
Americium by Rock@
Americium Contamination near a Transuranic Storage Area
Americium Soil Penetration at 230-S Building Crib Site@
Americium 241 on Two Subsoils from Differing Weathering
Americium 241, Neptunium 237, and Technetium 99
Americium@ Studies of Strontium, Technetium, Cesium,
Americium/Curium Separated from High-Level Reprocessing
Americium, and Curium in Soil-Aquatic Environments@
Analogues for Radioactive Waste Forms@
Analysis of Savannah River Plant High-Level Waste
Analysis@ Impact Statement: Radioactive Waste
Analysis@ Statement: Radioactive Waste Isolation in
Analysis and Development of Thermal Loading Guidelines@
Analysis and Evaluation of the Rock Mechanics Aspects
Analysis and Preliminary Results for Spherical Cavities@
Analysis and Program Plan@
Analysis for a Nuclear Waste Repository@
Analysis for the Nuclear Fuel Cycle-Parts 1 and 2:
Analysis for WIPP Repository@ @The
Analysis of the Back End of the Nuclear Fuel Cycle with
Analysis of the Fixation of High-Level Radioactive
Analysis of the Geological Stability of a Hypothetical
Analysis of the Measured Values for the State of Stress
Analysis of Combined Effects@
Analysis of Hydrologic Conditions Near the Site of a
Analysis of Plutonium-Release Consequences on the
Analysis of Salt Cavities Formed by Solution Mining: 1.
Analysis of Separation Methods Applied to Radioactive
Analysis of Shipping Containers@An Assessment of Stress-
Analysis of Soil Samples from Test Trench at Western
Analysis of Some Tracer Dyes in Granite Rock Using a
Analysis of Waste-Container Sleeve: 1. Initial
Analysis of Waste-Container Sleeve: 2. Influence of
Analysis of Waste-Container Sleeve: 4. Air Gap
Analysis Methods for Use in Assessing the Safety of the
Analysis Study for Waste Management Criteria@
Analysis@
Analysis@
Analysis@ of Spent Nuclear Fuel and Final Storage
Anhydrite and Brine and Their Implications for the
Anhydrite as a Nuclear Waste Repository Medium@
Animals@ of Microbial Processes to the Fate and
Annotated Bibliography@
Annotated Bibliography@ the Migration of Radionuclides
Annual Report of Lawrence Berkeley Laboratory Contract
Annual Report on the Characterization of High-Level
Annual Report on DOWT PROJECT AB0115A, FY-1975@
Annual Report October 1977 to September 1978
Annual Report@
Annual Report, November 1976-October 1977@
AB0115A, FY-1975@
Answer to Nuclear Waste Disposal?@
Antarctica - A Potential International Burial Area for
Antarctica@ Radioactive Waste on Ice - A Proposal
Antarctica, A Potential Disposal Site for the World's
Antarctica: Some Glaciological Aspects@
Anticline, Grand County, Utah@ @Subsurface Geology
Appalachian Basin in New York@ for Disposal of
Appalachian Basin with Reference to the Subsurface
Appalachian Mountain Region of Eastern United States
Applicability of Micromatrix Chromatography to Sorption
Appraisal of Hard Rock for Potential Underground
Appraisal of Hard Rock for Potential Underground
Appraisal of Underground Radioactive Waste Disposal in
April Through June 1976@ @Nuclear Waste Management
April 1 to September 30, 1977@ Analysis for the Nuclear
April 1-June 30, 1978@ of HLW-Wastes for the Sandia
April 1978@
April-September 1977)@ @Waste
Aquatic Environments@ Ratios and Diffusion Coefficients
Aqueous Radioactive Wastes@
Aqueous Waste Solidification Method@
Aqueous Wastes from Fuel Reprocessing Plants@
Aquifer System Below the Zone of Fresh Water, Atlantic
Aquifer. 1. Numerical Study of Convective-Diffusive
Aquifer. 2. Experimental Study of Convective-Diffusive
Aquifer. 3. Numerical Simulation of Brine Invasion of
Aquifers in the Southern Part of the Atlantic Coastal
Aquifers. Part 1 -- Development, Part 2 -- Documentation
Aquifers@ Brine Invasion into Fresh Water Aquifer. 3.
Area for High-Level Radioactive Wastes@
Area in Southeastern Idaho@ @Plutonium
Area 6, Technical Area 54@ @Geologic Description
Area 11@ @Plutonium Distribution
Area 17, Nevada Test Site, Using the Transient Method@

from Holes P-3 MP-1 through P-3 MP-5, Area 6, Technical	Area 5* *Geologic Description of Cores	000196
*Mine Renovation and Preparation of Experimental	Area*	000395
Waste Disposal Studies, Chemical Processing Plant	Area* *Underground	000192
to Rodents Inhabiting a Radioactive Waste Receiving	Area* *Radiation Doses	000203
from Alternative Fuel Cycles* *Estimates of Relative	Areas for the Disposal in Bedded Salt of LWR Wastes	000050
*Paleozoic Stratigraphy of Two	Areas in Southwestern Indiana*	000393
States Subsoils from Differing Weathering Intensity	Areas* 237, and Technetium 99 Sorption on Two United	000293
241 on Two Subsoils from Differing Weathering Intensity	Areas* of Technetium 99, Neptunium 237, and Americium	000294
Appraisal of Underground Radioactive Waste Disposal in	Argillaceous and Crystalline Rocks: Some Geochemical,	000218
Transient Method* *Measuring the Permeability of Eleana	Argillite from Area 17, Nevada Test Site, Using the	000040
*Studies of the Eleana	Argillite*	000423
of Plutonium 239, 240 Concentrations in a Stream near	Argonne National Laboratory* *A Preliminary Assessment	000201
Carcinogenic Hazard from Low-Level, Low-Rate *The	Argonne Radiological Impact Program (ARIP), Part I.	000252
Rate Radiation*The Argonne Radiological Impact Program (	ARIP), Part I. Carcinogenic Hazard from Low-Level, Low-	000253
Waterproof Geometry of Salt Domes, An Important Safety	Aspect for their Exploitation for Ultimate Disposal of	000407
*The Feasibility of Incorporating Radioactive Wastes in	Asphalt or Polyethylene*	000596
Air* *Autonomous	Aspiration Unit for Sampling Radioactive Aerosols from	000182
Waste Canisters and Spent Fuels *Neutron Interrogator	Assay System for the Idaho Chemical Processing Plant	000138
on the Disposal of Radioactive Wastes in the	Asse Salt Mine* *Recent Results and Developments	000177
*Function and Licensing Status of the	Asse Test Storage Facility*	000097
*Interim Storage of Spent Fuel	Assemblies*	000125
Nuclear *Program Scenario Analysis Methods for Use in	Assessing the Safety of the Geologic Isolation of	000542
Applied to Radioactive Waste Management* *Technology	Assessment and Impact Analysis of Separation Methods	000601
Disposal of High-Level Waste on the Ocean Floor* *	Assessment of the Radiological Protection Aspects of	000261
Disposal of High-Level Waste in Geologic *Preliminary	Assessment of the Radiological Protection Aspects of	000264
*Preliminary Safety	Assessment of the Waste Isolation Pilot Plant F ility*	000498
Radioactive *The United States Program for the Safety	Assessment of Geologic Disposal of Commercial	000319
*Long-Term Safety	Assessment of Geologic Isolation Repositories*	000269
Stream near Argonne National Laboratory* *A Preliminary	Assessment of Plutonium 239, 240 Concentrations in a	000201
Formations* *Long-Term Risk	Assessment of Radioactive Waste Disposal in Geologic	000256
Element Elastic-Plastic Analysis of Shipping *An	Assessment of Stress-Strain Data Suitable for Finite-	000143
Regarding Realistic Modeling that Allows Objective Risk	Assessment to be Made* and Ocean Disposal Concepts	000517
Volume 2. *Development and Application of a Risk	Assessment Method for Radioactive Waste Management.	000278
Associated with Radioactive Waste Management* *An	Assessment Methodology of Environmental Risks	000259
Laboratory Contract No. 45901AK. Waste Isolation Safety	Assessment Program - Collection and Generation of	000219
Characterization *Overview of Waste Isolation Safety	Assessment Program and Description of Source Term	000315
*Environmental	Assessment Related to Hanford Radioactive Waste Burial*	000268
Regulatory Commission* *Risk	Assessment Review Group Report to the U.S. Nuclear	000512
Elements and Radionuclides for Waste Management	Assessment* *Selection of Significant	000703
Site in New *Interaction of Radionuclides with Geomedia	Associated with the Waste Isolation Pilot Plant (WIPP)	000281
*Radionuclide Migration Studies	Associated with the WIPP Site in Southern New Mexico*	000280
*Rock Stresses	Associated with Burial of Nuclear Waste*	000085
*An Assessment Methodology of Environmental Risks	Associated with Radioactive Waste Management*	000259
Mesozoic Aquifer System Below the Zone of Fresh Water,	Atlantic Coastal Plain, North Carolina through New	000366
of Mesozoic Aquifers in the Southern Part of the	Atlantic Coastal Plain, South Carolina and Georgia*	000365
Radioactive *A Critical Study on the International	Atomic Energy Agency (IAEA) Definition of High-Level	000188
Characteristics of the Hanford Reservation of the U.S.	Atomic Energy Commission, Washington* and Ground-Water	000189
*Processing of High-Level	Atomic Wastes with the View to Ultimate Disposal*	000609
*Gas Generation from Radiolytic	Attack of Transuranic-Contaminated Hydrogenous Waste*	000091
*NWTS In-Situ Test Needs Workshop,	August 29-30, 1978*	000302
*Tennessee Valley	Authority Away-From-Reactor Spent Fuel Storage Studies*	000320
Plutonium in Waste* *An	Automatic Drum Scanning System for the Measurement of	000106
Aerosols from Air* *	Autonomous Aspiration Unit for Sampling Radioactive	000182
*Tennessee Valley Authority	Away-From-Reactor Spent Fuel Storage Studies*	000320
Spent Fuel Storage Requirements - The Need for	Away-From-Reactor Storage*	000705
*Geology of the 241-	AX Tank Farm*	000193
*Geology of the 241-	B Tank Farm*	000190
High-Level Waste Management. Volume *An Analysis of the	Back End of the Nuclear Fuel Cycle with Emphasis on	000321
*Transportation of Spent Fuel by	Barge and By Rail: A Comparison of Risk*	000545
I. Initial Studies of Strontium, Technetium, Cesium,	Barium, Cerium, Europium, Uranium, Plutonium, and	000028
Radioactive Wastes* *Glass as the First	Barrier for Long-Term Storage of Highly Active	000128
*Radioactive Waste Management: The Need for Multiple	Barriers*	000150
Beds in the Subsurface in Russell, Lincoln, Ellsworth,	Barton, and Rice Counties, Central Kansas* Part 1: Salt	000479
Porters Creek Clay, and Yazoo Clay in Parts of *	Basal Ottawa Limestone, Chattanooga Shale, Floyd Shale,	000429
*Thermal Properties of Gable Mountain	Basalt Cores Hanford Nuclear Reservation*	000053
*	Basalt Waste Isolation Program Annual Report*	000451
1978* *	Basalt Waste Isolation Program Monthly Report, April	000390
1978* *	Basalt Waste Isolation Program Monthly Report, February	000387
1978* *	Basalt Waste Isolation Program Monthly Report, January	000288
1978* *	Basalt Waste Isolation Program Monthly Report, March	000389
Sites of Repositories for Radioactive Wastes: Flood	Basalt* *Hypothetical Prototype	000488
Volume 7A. Repository Preconceptual Design Studies:	Basalt* Waste Isolation in Geologic Formations.	000589
Formations. Volume 7. Baseline Rock Properties -	Basalt* (GEIS): Radioactive Waste Isolation in Geologic	000023
Formations. Stratigraphics of Salt, Granite, Shale, and	Basalt* (GEIS): Radioactive Waste Isolation in Geologic	000385
of Storing Radioactive Wastes in Columbia River	Basalts* *The Possibility	000406
Technology* *Information	Base for Waste Repository Design, Volume 6. Excavation	000583
Waste Isolation in Geologic Formations. Volume 7.	Baseline Rock Properties - Basalt* (GEIS): Radioactive	000023
Waste Isolation in Geologic Formations. Volume 6.	Baseline Rock Properties - Shale* (GEIS): Radioactive	000022
; Radioactive Waste Isolation in Geologic Formations.	Baseline Rock Properties--Granite* Statement (GEIS)	000021
Waste Isolation in Geologic Formations. Volume 4.	Baseline Rock Properties: Salt* (GEIS): Radioactive	000020
*Potential of Denver	Basin for Disposal of Liquid Wastes*	000401
in Subsurface Rocks on North Flank of Appalachian	Basin in New York* for Disposal of Industrial Wastes	000029
*Subsurface Waste-Disposal Potential in Salina	Basin of Kansas*	000398
of the Interior Salt Domes of Northeast Texas Salt Dome	Basin to Investigate Their Suitability for Possible	000424
Radioactive Waste *Geologic Summary of the Appalachian	Basin with Reference to the Subsurface Disposal of	000279
*Removal of Cesium and Strontium from Fuel Storage	Basin Water*	000487
into the Idaho Chemical Processing Plant Fuel Storage	Basin Water* *Primary Source of Radionuclides Leaking	000488
*Geology of Subsurface Waste Disposal in Michigan	Basin*	000364

PERMUTED TITLE INDEX

@Regional Geology of the Northeast Texas Salt Dome	Basin*	000412
@Leak Testing Fuel Stored in the ICPP Fuel Storage	Basin*	000631
of Certain Domes of the Northeast Texas Salt-Dome	Basin* @Geology	000352
Growth of Salt Domes in the Northeast Texas Salt Dome	Basin* @Genesis and	000432
@Sedimentary History and Economic Geology of San Juan	Basin, New Mexico and Colorado*	000443
@Subsurface Disposal in Geologic	Basins - A Study of Reservoir Strata*	000408
@Economic and Industrial Potential of Geologic	Basins and Reservoir Strata*	000399
Potential at the Proposed Waste Isolation Pilot Plant @	Batch Kd Measurements of Nuclides to Estimate Migration	000300
Volume 16. Repository Preconceptual Design Studies:	Battelle Pacific Northwest Laboratory Waste Forms in	000550
Waste Management Research and Development Program at	Battelle Pacific* @High-Level Radioactive	000648
of Systems for Handling Radioactive Ion Exchange Resin	Beds* @The Design	000670
@Engineered Storage for the Disposal of Spent-Fuel	Bearing Components from Nuclear Facilities and General	000094
Packaging, Transportation, and Disposal of Spent-Fuel	Bearing Hardware Resulting from Light Water Reactor	000651
Prerequisites and Practice for Using Deep Water-	Bearing Horizons for Disposal of Liquid Radioactive	000203
Facilities at Hanford* @Characterization of Actinide-	Bearing Sediments Underlying Liquid Waste Disposal	000290
@Waste Management Capabilities for Alpha	Bearing Wastes at the Los Alamos Scientific Laboratory*	000419
for Repositories of Solid High-Level and Alpha-	Bearing Wastes in Geological Formations*@Site Selection	000413
to Minimize, Segregate, and Account for Alpha-	Bearing Wastes* @Operational Techniques	000610
Defense Waste, July-December 1977* @Yeldiwd	Bed Incineration System for U.S. Department of Energy	000340
@Estimates of Relative Areas for the Disposal in	Bedded Salt of LWR Wastes from Alternative Fuel Cycles*	000050
of a Hypothetical Radioactive Waste Repository in a	Bedded Salt Formation* of the Geological Stability	000562
of a High-Level Radioactive Waste Repository in	Bedded Salt* for Simulating the Deformation Behavior	000020
@Waste-Rock Interactions and	Bedrock Reactions*	000089
@Nuclide Transport by Groundwater in Swedish	Bedrock*	000001
Oil and Gas Test Holes in Central Kansas, Part 1: Salt	Beds in the Subsurface in Russell, Lincoln, Ellsworth,	000479
@Devitrification	Behavior in a Zinc Borosilicate Nuclear Waste Glass*	000679
Numerical Model for Simulating the Deformation	Behavior of a High-Level Radioactive Waste Repository	000520
Environments* @	Behavior of Candidate Canister Materials in Deep Ocean	000140
@Illite: A Model Based on the Sorption	Behavior of Cesium*	000033
@Geochemical	Behavior of Long-Lived Radioactive Wastes*	000254
Compression up to 200 C* @Mechanical	Behavior of New Mexico Rock Salt in Triaxial	000006
@The	Behavior of Radioactive Fission Products in Soils*	000272
Water Disposal into a Limited @Contamination of Soils:	Behavior of Radioactive Iron in Unsaturated Soils after	000292
The Relationship of Microbial Processes to the Fate and	Behavior of Transuranic Elements in Soils, Plants and	000310
in Selected Rocks: 1977 Annual Report of Lawrence	Berkeley Laboratory Contract No. W7405ENG. Waste	000219
@R	Bibliography on Ocean Waste Disposal*	000465
@Radioactive Waste Disposal in Salt Deposits - A	Bibliography with Abstracts*	000036
@Ocean Waste Disposal (A	Bibliography with Abstracts)*	000367
@Decommissioning of Nuclear Facilities - An Annotated	Bibliography*	000326
Water at the Nevada Test Site - Part 2: Annotated	Bibliography* the Migration of Radionuclides in Ground	000163
@Corrosion Monitoring of Storage	Bins for Radioactive Calcines*	000116
@Feasibility Study of Underground Storage of	Bitumen Blocks Containing Radioactive Wastes*	000501
Temperature Effects in Case of Underground Storage of	Bitumen* @Radiolysis and	000202
Wastes from Nuclear Power Plants into Concrete and	Bitumen* @Incorporation of Radioactive	000653
Level Wastes and Spent Solvent and on Fixation into	Bitumen* @Recent Experiments on the Treatment of Iodine	000574
@Influence of Surfactants on	Bituminization of Radioactive Waste*	000626
Research Center Karlsruhe - Experience from Plant @	Bituminization of Radioactive Wastes at the Nuclear	000612
@Evaluation of the Activity Levels in Fusion Reactor	Blankets. Waste Problem*	000699
Leaching of Radioactivity from Highly Radioactive Glass	Blocks Buried Below the Water Table: Fifteen Years of	000645
@Feasibility Study of Underground Storage of Bitumen	Blocks Containing Radioactive Wastes*	000501
Report: Geologic Feasibility of Talc and Serpentinite	Bodies from the Appalachian Mountain Region of Eastern	000484
@	Bonding of Metal Oxides in Sodium Silicate Glass*	000126
Plutonium from High-Level Caustic Waste Solutions Using	Bone Char - Pilot Study* @Removal of	000570
Clayey Formation (Application to the Case of the	Boom Clay at Mol)* Migration in a Homogeneous	000291
Site* @	Borehole Drilling and Related Activities at the Strips	000422
@Preliminary Report on Geophysical and Mechanical	Borehole Measurements at Strips*	000437
Isolation of Radioactive Waste* @The States of	Borehole Plugging and Shaft Sealing for Geologic	000536
1. @Investigation of Salt Transport in Vertical	Boreholes and Brine Invasion into Fresh Water Aquifer.	000273
2. @Investigation of Salt Transport in Vertical	Boreholes and Brine Invasion into Fresh Water Aquifer.	000274
3. @Investigation of Salt Transport in Vertical	Boreholes and Brine Invasion into Fresh Water Aquifer.	000275
Progress Report for the @Comet Technology for Plugging	Boreholes in Radioactive Waste Repository Sites:	000059
Volumes 1 and 2* @Feasibility of Sealing	Boreholes with Compacted Natural Earthen Material,	000052
Volume 3* @Feasibility of Sealing	Boreholes with Compacted Natural Earthen Materials,	000066
@ @Effect of Internal Alpha Radiation on	Borosilicate Glass Containing Savannah River Plant Waste	000099
Radioactive Waste* @Volatility of a Zinc	Borosilicate Glass Containing Simulated High-Level	000604
of an Improved High Level Radioactive Waste (HLW)	Borosilicate Glass* @Preparation and Characterization	000605
Part 2. The Experimental Evaluation of Phosphate and	Borosilicate Glasses* of Radioactive Waste in Glass.	000100
@Devitrification Behavior in a Zinc	Borosilicate Nuclear Waste Glass*	000679
Coolants for the Transportation of Liquid Metal Fast	Breeder Reactor Spent Fuel Rods* @Evaluation of Organic	000096
Hydrite as a @P-T-X Relations of Hydrite and	Brine and Their Implications for the Suitability of	000016
@Corrosion of Metals in a Hot, High Pressure	Brine Environment*	000130
Gradients, Stress, and Irradiation on Migration of	Brine Inclusions in a Salt Repository* Temperature	000041
of Salt Transport in Vertical Boreholes and	Brine Invasion into Fresh Water Aquifer. 1. Numerical	000273
of Salt Transport in Vertical Boreholes and	Brine Invasion into Fresh Water Aquifer. 2.	000274
of Salt Transport in Vertical Boreholes and	Brine Invasion into Fresh Water Aquifer. 3. Numerical	000275
into Fresh Water Aquifer. 2. Numerical Simulation of	Brine Invasion of Fresh Water Aquifers* Brine Invasion	000275
@Hydrothermal Glass Reactions in Salt	Brine*	000155
@The Sandia Solidification Process - A	Broad Range Aqueous Waste Solidification Method*	000636
Wastes* @The	Brookhaven Experience with Sea Disposal of Radioactive	000598
@Plutonium-Bearing Soil Penetration at 238-b	Building Crib Sites*	000170
Determined by the Jenite and Johnson @Flowability of	Bulk Solids Used in Shale Fracturing Sports as	000641
@Thermal Conductivity of the Rocks in the	Bureau of Mines Standard Rock Suite*	000063
@Rock Stresses Associated with	Burial of Nuclear Waste*	000005
A Perspective on the Relative Hazard of Deep	Burial of Nuclear Wastes* in Deep Geologic Formations:	000008
@Suggested Concentration Limits for Shallow Land	Burial of Radionuclides*	000277
@Antarctica - A Potential International	Burial Area for High-Level Radioactive Wastes*	000352
@	Burial Grounds*	000168
@Development of Solidification and	Burial Methods for Radioactive Wastes of Fuel Cycle*	000392

PERMUTED TITLE INDEX

Its Alteration or Geological Occurrences and in Deep- Burial Radioactive Waste Disposal* @Pollucite and 000185
Between Processing Method and Radioactive Waste Burial Site* @Interaction 000063
Radioactive Waste @Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Low-Level 000167
Level Radioactive @Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low- 000166
Level Radioactive @Evaluation of Isotope Migration-Land Burial Water Chemistry at Commercially Operated Low- 000168
Management Related to Hanford Radioactive Waste Burial* @Environmental 000268
of Radioactivity from Highly Radioactive Glass Blocks Buried Below the Water Table: Fifteen Years of Results* 000645
Engineering Laboratory* @Retrieval of Buried Transuranic Waste at the Idaho National 000188
@Geology of the 201- BY Tank Farm* 000195
New Mexico Rock Salt in Triaxial Compression up to 200 C* @Mechanical Behavior of 000086
in the Nuclear Fuel Cycle - Conversion of Yellow Cake to Uranium Hexafluoride. Part 1. The Fluorination- 000521
Storage Facility at Idaho @Properties of Radioactive Calcine Retrieved from the Second Calcined Solids 000208
@Stabilization of Simulated Commercial Waste Calcine* 000660
of Radioactive Calcine Retrieved from the Second Calcined Solids Storage Facility at Idaho Chemical 000204
@Sampling of Stored High-Level Radioactive Calcined Waste at Idaho Chemical Processing Plant* 000154
@Vitrification of High-Level ICPP Calcined Waste* 000602
(2). Leachabilities of Cesium and Strontium from Calcined Zeolites* Waste Solution with Zeolite 000619
@Corrosion Monitoring of Storage Bins for Radioactive Calcines* 000116
Saline Aquifers. Part 1 — Development, @A Model for Calculating Effects of Liquid Waste Disposal in Deep 000267
Monitoring at the Waste Isolation Pilot Plant Site, Calendar Year 1976* @Environmental 000363
@Growth and Shape of Salt Cavity—Computer Model and Calibration from Detroit Mine Experiment* 000062
@ Canada Keeps Its Disposal Options Open* 000073
@Underground Disposal of Canada's Nuclear Waste* 000070
@The Management of Canada's Nuclear Wastes* 000317
and High-Level Wastes* @The Canadian Program for Storage and Disposal of Spent Fuel 000325
@Preliminary Corrosion Test of a Glass-Ceramic Candidate for a Nuclear Waste Canister* 000151
@Behavior of Candidate Canister Materials in Deep Ocean Environments* 000148
Isolation* @High-Level Waste Canister Corrosion Studies Pertinent to Geologic 000103
@High-Level Waste Canister Emplacement and Retrieval Concepts Study* 000561
@Behavior of Candidate Canister Materials in Deep Ocean Environments* 000148
@History of Prototype High Level Waste Canister SS-9 While in Air and Water Storage* 000101
@Nuclear Waste Canister Thermally Induced Motion* 000025
Test of a Glass-Ceramic Candidate for a Nuclear Waste Canister* @Preliminary Corrosion 000151
System for the Idaho Chemical Processing Plant Waste Canisters and Spent Fuels Preliminary Description and 000138
@Corrosion Resistance of Canisters for Final Disposal of Spent Nuclear Fuel* 000132
@Corrosion Considerations for Nuclear Waste Isolation Canisters* 000102
@Improved Heat Transfer from Radioactive Waste Canisters* 000119
Distribution in a Cavity and Chimney Produced by the Canmikin Nuclear Explosion, Amchitka Island, Alaska* 000377
Scientific Laboratory* @Waste Management Capabilities for Alpha Bearing Wastes at the Los Alamos 000419
@ @Waste Management Programs and Capabilities within the Los Alamos Scientific Laboratory 000471
@ Carbon 14 Production in Nuclear Reactors* 000598
@Methods Evaluation for the Continuous Monitoring of Carbon 14, Krypton 85 and Iodine 129 in Nuclear Fuel 000285
Storage Space in the Hidcentment, @Cursory Search for Carbonates Suitable for Developing Dry Underground 000371
The Argonne Radiological Impact Program (ARIP), Part I. Carcinogenic Hazard from Low-Level, Low-Rate Radiation* 000253
the Southern Part of the Atlantic Coastal Plain, South Carolina and Georgia* Potential of Mesozoic Aquifers in 000265
the Zone of Fresh Water, Atlantic Coastal Plain, North Carolina through New Jersey* Aquifer System Below 000266
@Mineral Exchange at the University of North Carolina* 000677
Waste at the Savannah River Plant, Aiken, South Carolina* Long-Term Management of Defense Transuranic 000286
@Development of High-Level Waste Shipping Cask Models* 000112
of Radiological Consequences from Sabotage of Shipping Casks for Spent Fuel and High-Level Waste*@Calculations 000506
@Fire Protection Device for LWBR Spent Fuel Shipping Casks* 000568
Reactivity of Simulated Nuclear Waste Forms and Water- Catalyzed Waste-Rock Interactions* @Hydrothermal 000058
@Sorption and Fixation of Large Cations by Shale Formations* 000018
Linde 13X Equilibrium with Trivalent Cerium and Yttrium Cations* @ 000002
@Removal of Plutonium from High-Level Caustic Waste Solutions Using Bone Char - Pilot Study* 000578
Program on Radioactive Waste Storage in Bined Caverns in Crystalline Rock, Program Summary* 000487
Analysis and Preliminary @Structural Analysis of Salt Cavities Formed by Solution Mining: 1. Method of 000540
of Analysis and Preliminary Results for Spherical Cavities* Cavities Formed by Solution Mining: 1. Method 000540
Hydrologic Processes and Radionuclide Distribution in a Cavity and Chimney Produced by the Canmikin Nuclear 000377
Mine Experiment* @Growth and Shape of Salt Cavity—Computer Model and Calibration from Detroit 000062
Fuel Handling and Packaging Program. A Survey of Hot Cell Facilities* @Spent 000135
90 and Cobalt 60 Leached from Matrix of Hardened Cement Grout* on Amounts of Cesium 137, Strontium 000639
@Measurement and Control of Cement Set Times in Waste Solidification* 000678
Waste Repository Sites: Progress Report for the Period @ Cement Technology for Plugging Boreholes in Radioactive 000059
@Permeability Measurements on Cementitious Materials for Nuclear Waste Isolation* 000156
@Role of Admixtures in Preparing Dense Cements for Radioactive Waste Isolation* 000642
Intermediate-Level Liquid Wastes by Solidification in Cements* @Immobilization of Strontium and Cesium in 000668
of Radioactive Wastes at the Nuclear Research Center Karlsruhe - Experience from Plant Operation and 000612
@Western New York Nuclear Service Center Study, Final Report for Public Comment* 000211
from Test Trench at Western New York Nuclear Service Center, West Valley, New York* Analysis of Soil Samples 000173
Russell, Lipscomb, Ellsworth, Barton, and Rice Counties, Central Kansas* Part 1: Salt Beds in the Subsurface in 000479
Russell, @Salt Dissolution in Oil and Gas Test Holes in Central Kansas, Part 1: Salt Beds in the Subsurface in 000479
@Characterization of a Titanate Based Ceramic for High Level Nuclear Waste Solidification* 000122
@Preliminary Corrosion Test of a Glass- Ceramic Candidate for a Nuclear Waste Canister* 000151
Radioactive Operation Experience with a Joule Heated Ceramic Melter for Vitrification of High-Level Liquid 000686
@Characterization of Glass and Glass Ceramic Nuclear Waste Forms* 000635
@Metal Encapsulation of Ceramic Nuclear* 000618
Level Waste Products* @Clay-Phosphate Ceramics and Vitrocerts: Alternatives to Monolithic High 000689
@Linde 13X Equilibrium with Trivalent Cerium and Yttrium Cations* 000002
@Migration of Strontium 90 and Cesium 137 in Soils of Different Textures* 000227
Studies of Strontium, Technetium, Cesium, Barium, Cerium, Europium, Uranium, Plutonium, and Americium* 000028
@ Ceramic for High Level Waste Containment* 000569
Waste Solution with Zeolite (2). Leachabilities of Cesium and Strontium from Calcined Zeolites*Radioactive 000619
@Removal of Cesium and Strontium from Pool Storage Basin Water* 000687
@Investigations of the Volatility of Cesium from Aluminosilicates by Mass Spectrometry* 000138
Solidification in @Immobilization of Strontium and Cesium in Intermediate-Level Liquid Waste by 000668
@The Sorption of Microquantities of Strontium and Cesium in Soils* 000045
@Mass Spectra of Cesium Aluminosilicate Waste Forms* 000113

PERMUTED TITLE INDEX

$Evaluation of Matrix $Effect of Dimension of Specimen on Amounts of	Cesium 137 Sorbents for Fixation in Concrete* 000655
$Illite: A Model Based on the Sorption Behavior of	Cesium 137, Strontium 90 and Cobalt 60 Leached from 000639
Exchange Equilibria with Strontium-Sodium-	Cesium* 000033
Granite. 1. Initial Studies of Strontium, Technetium,	Cesium* $Montmorillonite 000027
$Polonium and Plutonium in An Intertidal Food	Cesium, Barium, Cerium, Europium, Uranium, Plutonium, 000028
Waste* $Technical	Chain* 000235
from High-Level Caustic Waste Solutions Using Bone	Challenges in the Immobilization of Hanford Defense 000627
Level Nuclear Waste Solidification* $	Char - Pilot Study* $Removal of Plutonium 000578
Waste (HAW) Borosilicate Glass* $Preparation and	Characterization of a Titanate Based Ceramic for High 000122
Underlying Liquid Waste Disposal Facilities at Hanford* $	Characterization of an Improved High Level Radioactive 000605
Waste Forms* $	Characterization of Actinide-Bearing Sediments 000290
$Annual Report on the	Characterization of Glass and Glass Ceramic Nuclear 000635
High-Level Waste Glasses by Sputter-Induced Optical $	Characterization of High-Level Waste Glass 000661
$Leach Rate	Characterization of Leached Surface Layers on Simulated 000118
Products Formed with Radioactive $Physicochemical	Characterization of Solid Radioactive Waste Forms* 000246
Wasteforms* $Microstructural	Characterization of Solidification Agents used and 000621
WIPP Site, Southeastern New Mexico. $Geological	Characterization of Solidified Simulated Reactor 000115
WIPP Site, Southeastern New Mexico. $Geological	Characterization Report, Waste Isolation Pilot Plant (000073
WIPP Site, Southeastern New Mexico* $Geological	Characterization Report, Waste Isolation Pilot Plant (000074
WIPP Site, Southeastern New Mexico. $Geological	Characterization Report, Waste Isolation Pilot Plant (000035
WIPP Site, Southeastern New Mexico. Site $Geological	Characterization Report, Waste Isolation Pilot Plant (000053
WIPP Site, Southeastern New Mexico. $Geological	Characterization Report, Waste Isolation Pilot Plant (000054
WIPP Site, Southeastern New Mexico. $Geological	Characterization Report, Waste Isolation Pilot Plant (000055
Assessment Program and Description of Source Term	Characterization Report, Waste Isolation Pilot Plant (000056
for High-Level Radioactive Waste Repository Site	Characterization Task at Pacific Northwest Laboratories* 000315
Chemical Testing* $Low-Level Solidified Waste	Characterization* $An Evaluation of Kriging Techniques 000026
Yazoo Clay in Parts of $Basal Ottawa Limestone.	Characterization: The Scientific Basis for Physical and 000003
Leach Fuel Hulls* $	Chattanooga Shale, Floyd Shale, Porters Creek Clay, and 000030
Progress Report, January-March 1978* $	Chemical Decontamination and Melt Densification of Chop- 000588
$Underground Waste Disposal Studies,	Chemical Engineering Division Fuel Cycle Programs 000336
$Primary Source of Radionuclides Leaking into the Idaho	Chemical Processing Plant Area* 000192
$Corrosion Evaluation of Stainless Steels in Idaho	Chemical Processing Plant Fuel Storage Basin Water* 000188
Fuels $Neutron Interrogator Assay System for the Idaho	Chemical Processing Plant High Level Radioactive Waste 000117
of a Pelleted Waste Form for High-Level Idaho	Chemical Processing Plant Waste Canisters and Spent 000138
$Reprocessing Facilities, Idaho	Chemical Processing Plant Zirconia Wastes* $Development 000628
Stored High-Level Radioactive Calcined Waste at Idaho	Chemical Processing Plant* 000583
the Second Calcined Solids Storage Facility at Idaho	Chemical Processing Plant* $Sampling of 000158
of Defense High-Level Radioactive Waste Idaho	Chemical Processing Plant* Calcine Retrieved from 000204
Waste from Light-Water Reactors and Physical and	Chemical Processing Plant, Idaho Falls, Idaho* 000346
$United States Geological Survey Studies of Physical-	Chemical Properties* $Sources of Radioactive 000691
$Importance of the	Chemical Relationships in Salt Repositories* 000079
Characterization: The Scientific Basis for Physical and	Chemical Species Formed in the Interaction* 000032
Radionuclide Mobility and Retention; Element	Chemical Testing* $Low-Level Solidified Waste 000003
$Evaluation of Isotope Migration-Land Burial Water	Chemistry and Geochemistry; Conclusions and Evaluation* 000217
$Evaluation of Isotope Migration-Land Burial Water	Chemistry at Commercially Low-Level Radioactive Waste 000167
$Evaluation of Isotope Migration-Land Burial Water	Chemistry at Commercially Operated Low-Level 000166
Temperature and Pressure to $Application of Physical	Chemistry at Commercially Operated Low-Level 000[illegible]
and Potential of Prediction* $	Chemistry of Fluids in Rock Salt at Elevated 000080
with Unweathered Igneous Rocks* $	Chemistry of Nuclear Waste Glass Reactions: Problems 000093
Processes and Radionuclide Distribution in a Cavity and	Chemistry of Technetium 99 and Neptunium 237 in Contact 000228
$Chemical Decontamination and Melt Densification of	Chimney Produced by the Cannikin Nuclear Explosion, 000377
$Problems in the Disposal of Irradiated Fuel	Chop-Leach Fuel Hulls* 000588
Investigation of the Virgin River Valley Salt Deposits,	Cladding* 000416
Clayey Formation (Application to the Case of the Boom	Clark County, Southeastern Nevada to Investigate their 000439
Shale, Floyd Shale, Porters Creek Clay, and Yazoo	Clay at Moll)*of Radionuclide Migration in a Homogeneous 000291
$Sorption and Retention by	Clay in Parts of Alabama, Mississippi and Tennessee as 000030
$Fixation of Radioactive Wastes in	Clay Minerals: Oak Ridge Studies* 000675
Experiments for Self Fixation of Radioactive Wastes in	Clay-Flux Mixes* 000682
Monolithic High Level Waste Products $	Clay-Flux Mixtures* $Heat 000617
Chattanooga Shale, Floyd Shale, Porters Creek	Clay-Phosphate Ceramics and Vitrosets: Alternatives to 000689
Modelling of Radionuclide Migration in a Homogeneous	Clay, and Yazoo Clay in Parts of Alabama, Mississippi 000030
$Studies on Fixation of Radioactivity in Fired	Clayey Formation (Application to the Case of the Boom 000291
of Radioactive Waste by Hydrothermal Reactions with	Clays* 000678
$Heater Experiments in the	Clays* $Fixation 000575
$Some Properties of Hydrogen-Based	Climax Stock, Nevada Test Site* 000446
Reprocessing Plant at West $Issues Related to the	Clinoptilolite* 000638
of Waste-Container Sleeve: 4. Air Gap Influence on Hole	Closing of the Nuclear Fuel Services, Incorporated, 000169
$Oceanographic Research on Waste Disposal Off the	Closure* $Thermoelastic/Plastic Analysis 000141
$Geohydrologic Studies of Gulf	Coast of Tokai-Mura* 000475
Radioactive $An Investigation of the Utility of Gulf	Coast Interior Salt Domes* 000460
Aquifer System Below the Zone of Fresh Water, Atlantic	Coast Salt Domes for the Storage or Disposal of 000428
Mesozoic Aquifers in the Southern Part of the Atlantic	Coastal Plain, North Carolina through New Jersey* 000366
of Specimen on Amounts of Cesium 137, Strontium 90 and	Coastal Plain, South Carolina and Georgia* Potential of 000365
$Determination of Distribution Ratios and Diffusion	Cobalt 60 Leached from Matrix of Hardened Cement Grout* 000639
$Investigation of Cooling	Coefficients of Neptunium, Americium, and Curium in 000301
85901AK. Waste Isolation Safety Assessment Program -	Coil Corrosion in Radioactive-Waste Storage Tanks* 000139
and Economic Geology of San Juan Basin, New Mexico and	Collection and Generation of Transport Data* No. 000219
$The Feasibility of Storing Radioactive Wastes in	Colorado* $Sedimentary History 000043
Waste Disposal Report on Seepage Pit Liquid Waste-Shale	Columbia River Basalts* 000436
$Management of	Column Experiment* $Radioactive 000178
$Nuclear Waste Management--Review of	Commercial High-Level Radioactive Waste* 000551
$The Management and Storage of	Commercial Nuclear Waste Management* 000610
for the Safety Assessment of Geologic Disposal of	Commercial Power Reactor Wastes* 000681
$Description of Solid Waste Expected from a Large	Commercial Radioactive Wastes*The United States Program 000319
$Stabilization of Simulated	Commercial Reprocessing Plant* 000702
to Draft Generic Environmental Impact Statement on	Commercial Waste Calcine* 000660
of Isotope Migration-Land Burial Water Chemistry at	Commercial Waste Management: Radioactive Waste 00065
	Commercially Low-Level Radioactive Waste Disposal Sites* 000167

of Isotope Migration-Land Burial Water Chemistry at
of Isotope Migration-Land Burial Water Chemistry at
#Nuclear Regulatory
Review Group Report to the U.S. Nuclear Regulatory
of the Hanford Reservation of the U.S. Atomic Energy
#Feasibility of Sealing Boreholes with
#Feasibility of Sealing Boreholes with
#
#The Radioactive Waste Management
Review Document for the Radioactive Waste Management
#Retention of Organic-
the Adsorption of Radiostrontium in Saclay Soil. Three-
Engineered Storage for the Disposal of Non-Fuel Bearing
Contaminated Soil as a Function of pH and Certain Soil
#Vitrification of High Sodium-Aluminum Wastes:
Waste#
#Development of Glass
#Development of Glass
Permeability#
#
Mechanical Behavior of New Mexico Rock Salt in Triaxial
Nuclear Waste Repository#
#
Experiment#
#Growth and Shape of Salt Cavity--
#Thermophysical Properties of
Level Reprocessing Waste#
#Final Purification and
#Putting Evaporators to Work:
Radionuclides#
#Suggested
#A Preliminary Assessment of Plutonium 239, 240
States and in Deer from an
#Comparison of Plutonium
#Revised
Waste in Fine-Grained Deposits#
#A
Using Underground Mining Techniques for
#Preliminary
#Comparisons Among Alternatives,
#Comparison of Some Geologic and Ocean Disposal
#High-Level Waste Canister Emplacement and Retrieval
General Nuclear Services On-Site Solid Waste Storage
#Design Aspects of the Alpha Repository: 2.
and Retention; Element Chemistry and Geochemistry;
Nuclear Reactor Concerning the Evolution in the Ground #
of Radioactive Wastes from Nuclear Power Plants into
Waste#
#Studies of
River Plant Waste#
#Evaluation of
#Radiolytic Gas Production from
#Evaluation of Cesium 137 Sorbents for Fixation in
of the Fixation of High-Level Radioactive Wastes in
FUSTAR (Formed Under Elevated Temperature and Pressure)
Standard Rock Suite#
#Thermal
#Thermal
Summary#
#National Waste Terminal Storage Program,
#
#The Utilization of the HOL Data Structure in
#Hot Isostatic Pressing for the
from Stored Nuclear Waste in Natural Rock
#Geochemical
Waste Isolation in Geologic Formations. Facility
the Sleeve#
#Thermoelastic/Plastic Analysis of Waste-
Sleeve Loading#
#Thermoelastic/Plastic Analysis of Waste-
#Thermoelastic/Plastic Analysis of Waste-
#Effects of Radiations from Transuranium Nuclides on
#Durability of
#Maximum Acceptable Temperatures of Wastes and
for Finite-Element Plastic-Plastic Analysis of Shipping
#Hot Isostatic Pressing for the Consolidation and
#The Distribution of Plutonium in a Rock
#Potential Mechanisms of
#Cements for High Level Waste
September 1978 Mathematical Simulation of Sediment and
#Disposition (D and D) of Retired
#Gas Generation from Radiolytic Attack of Transuranic-
#
#Extractability of Plutonium 239 and Curium 242 from a
#Waste Management of Actinide
and Long-Term Problems of High-Level and Actinide-
#Radiolytic Gas Generation in Plutonium
Southeastern Idaho#
#Plutonium and Americium
Unsaturated Soils after Water Disposal into a Limited #
1977 Annual Report of Lawrence Berkeley Laboratory
Isolation Repository#
#An Overview of Material
Isolation Repository#
#Material
#Experience in the Organization of Sanitary Radiometric
#Measurement and
Elements by Geologic Media##Electron Transfer Method of
Sources in Porous Media#
#Steady Thermal
into Fresh Water Aquifer. 1. Numerical Study of
into Fresh Water Aquifer. 2. Experimental Study of
Impact of Waste Effluents in the Nuclear Fuel Cycle -
Breeder Reactor Spent Fuel Rods# #Evaluation of Organic
Fuel Cycle - Reprocessing of High-Temperature Gas-
#Forced-Air
Tanks#
#Investigation of
Storage Tanks#
#Heat Pipe

Commercially Operated Low-Level Radioactive Waste 000166
Commercially Operated Low-Level Radioactive Waste 000168
Commission Program for High Level Waste Management# 000508
Commission#
#Risk Assessment 000512
Commission, Washington#and Ground-Water Characteristics 000169
Compacted Natural Earthen Material, Volumes 1 and 2# 000052
Compacted Natural Earthen Materials, Volume 3# 000066
Compilation of Hanford Corrosion Studies# 000129
Complex and Its Operations# 000208
Complex at the Idaho National Engineering Laboratory# 000176
Complexed Plutonium in Soil# 000295
Component System#
#Preliminary Investigations of 000165
Components from Nuclear Facilities and General 000098
Components: HNO3-NaOH System# 238 and Curium 242 from a 000288
Composition Ranges and Properties# 000104
Compositions for Immobilization of Savannah River Plant 000656
Compositions for Immobilization of SRP Waste# 000656
Compressible Fluid Flow Through Rocks of Variable 000049
Compression up to 200 C#
000096
Computer Enhanced "Release Scenario" Analysis for a 000560
Computer Model and Calibration from Detroit Mine 000462
Conasauga Shale# 000076
Concentration of Americium/Curium Separated from High- 000409
Concentration of Aqueous Radioactive Waste# 000603
Concentration Limits for Shallow Land Burial of 000277
Concentrations in a Stream near Argonne National 000201
Concentrations in Deer from the Southeastern United 000271
Concept for the Waste Isolation Pilot Plant# 000537
Concept for Long-Term Isolation of Solid Radioactive 000376
Concept of Waste Retrieval System-Tank Wall Access 000630
Concepts and Modules# 000345
Concepts Regarding Realistic Modeling that Allows 000517
Concepts Study# 000561
Concepts#
#Description of Allied- 000205
Conceptual Layouts of Underground Storage Facilities # 000541
Conclusions and Evaluation#
Radionuclide Mobility 000217
Conclusions Drawn from Studies on the Oklo Natural 000287
Concrete and Bitumen#
#Incorporation 000653
Concrete as a Host for Savannah River Plant Radioactive 000672
Concrete as a Matrix for Solidification of Savannah 000467
Concrete Containing Savannah River Plant Waste# 000098
Concrete# 000655
Concrete# Preliminary Engineering and Economic Analysis 000695
Concretes - Experimental Program and Initial Results#in 000060
Conductivity of the Rocks in the Bureau of Mines 000063
Conductivity of Rock Salt from Louisiana Salt Domes# 000061
Configuration Management Plan, Volume 1. Management 000499
Confinement of Volatilized Ruthenium Oxides# 000637
Consequence Analysis for WIPP Repository# 000223
Consolidation and Containment of Radioactive Waste# 000629
Constraints on Accumulation of Actinide Critical Masses 000015
Construction Feasibility and Costs by Rock Type## 000586
Container Sleeve: 1. Initial Estimates of Loading on 000067
Container Sleeve: 3. Influence of Salt Strength on 000080
Container Sleeve: 4. Air Gap Influence on Hole Closure# 000180
Container Surfaces# 000123
Containers for Storing Solidified Radioactive Wastes# 000095
Containers During Retrievable Geologic Storage# 000121
Containers#An Assessment of Stress-Strain Data Suitable 000183
Containment of Radioactive Waste# 000629
Containment Environment# 000031
Containment Failure# 000558
Containment# 000569
Contaminant Transport in Surface Waters#October 1977 to 000289
Contaminated Facilities at Hanford# 000175
Contaminated Hydrogenous Waste# 000051
Contaminated Reactor Facilities# 000692
Contaminated Soil as a Function of pH and Certain Soil 000288
Contaminated Soil# 000688
Contaminated Spent Fuel Reprocessing Wastes from the 000035
Contaminated Waste Materials# 000083
Contamination near a Transuranic Storage Area in 000181
Contamination of Soils: Behavior of Radioactive Ions in 000292
Contract No. 85901AK. Waste Isolation Safety Assessment 000219
Control and Accountability Considerations for a Waste 000896
Control and Accountability Procedures for a Waste 000120
Control at the Stage of Transporting Solid Radioactive 000567
Control of Cement Set Times in Waste Solidification# 000678
Controlling Eh During Adsorption of Multivalent 000070
Convection at Low Rayleigh Number from Concentrated 000077
Convective-Diffusive Flow in Wellbores# Brine Invasion 000273
Convective-Diffusive Flow in Wellbores# Brine Invasion 000274
Conversion of Yellow Cake to Uranium Hexafluoride. Part 000521
Coolants for the Transportation of Liquid Metal Fast 000096
Cooled Reactor Fuel Containing Uranium 233 and Thorium# 000501
Cooling of a Waste Isolation Pilot Plant Mine Drift# 000532
Cooling Coil Corrosion in Radioactive-Waste Storage 000139
Cooling System for Underground, Radioactive Waste 000107

PERMUTED TITLE INDEX

Disposal* *Organization for Economic
Mined Caverns in Crystalline Rock, *Swedish-American
Technical Area 5B* *Geologic Description of
*Thermal Properties of Gable Mountain Basalt
Treatment* *
the Environmental Impact of Waste Effluents in the *
the Environmental Impact of Waste Effluents in the *
the Environmental Impact of Waste Effluents in the *
Microstructures of Glasses Containing Savannah River
*Investigation of Cooling Coil
Environment* *
Canisters* *
Chemical Processing Plant High Level Radioactive Waste *
Calcines* *
Spent Nuclear Fuel* *
*High-Level Waste Canister
*Compilation of Hanford
Nuclear Waste Canister* *Preliminary
*Correlation of Environmental Impact with
*Nuclear Shipping and Waste Disposal
in the *Correlation of Radioactive Waste Treatments
in the *Correlation of Radioactive Waste Treatment
in the *Correlation of Radioactive Waste Treatment
For* *i s. Facility Construction Feasibility and
*
*
in Nass. , Lincoln, Ellsworth, Barton, and Rice
of a Potential Nuclear Waste Repository, Eddy and Lea
*Organizatics for Economic Cooperation and Development
of the Virgin River Valley Salt Deposits, Clark
Waste Replacement Site, Salt Valley Anticline, Grand
Limestone, Chattanooga Shale, Floyd Shale, Porters
*Plutonium-Americium Soil Penetration at 238-S Building
*Review of
*Geological
*Considerations of Environmental Protection
Wastes* *Geological
*Systems Analysis Study for Waste Management
National Waste Terminal Storage Program Waste Packaging
*Geochemical Constraints on Accumulation of Actinide
Ecosystems* *
Agency (IAEA) Definition of High-Level Radioactive *A
Waste Incinerator* *Selection of a
Measured Values for the State of Stress in the Earth's
Term Geochemical Interactions of High Active Waste with
*Fracture Detection in
on Radioactive Waste Storage in Mined Caverns in
*In-Situ Experiments on Nuclide Migration in Fractured
Radioactive Waste Disposal in Argillaceous and
and Diffusion Coefficients of Neptunium, Americium, and
*Final Purification and Concentration of Americium/
and Certain Soil *Extractability of Plutonium 238 and
*
*Environmental Effects from
on the Public* *Effects of
Dry Underground Storage Space in the Midcontinent, U.S.*
from the Uranium-Plutonium and Thorium-Uranium Fuel
in Bedded Salt of LWR Wastes from Alternative Fuel
September, 1978* *TRU Waste
*Disposition (
Utilizing Measurements Made During Electron *Disposition (D and
Waste Disposal Applications* *Radiation
* *Radiation
*Helium Formation from Alpha-
in the Waste Isolation Pilot Plant (WIPP). October 1-
in the Waste Isolation Pilot Plant (WIPP), October 1-
Expected in the Waste Isolation Pilot Plant (WIPP).
Quarterly Progress Report, October Through
Reprocessing Division of the Department of Energy (July-
for U.S. Department of Energy Defense Waste, July-
Liquid TRU Waste from the Nuclear Fuel Cycle: October-
Waste Management in the United States: A Time for
Reprocessing Plant* *Management of Highly Active
Bibliography* *
*
Engineering Laboratory* *Decontamination and
Idaho National Engineering Laboratory* *
Fuel Wells* *Chemical
*Ion Exchange
Formations: A Perspective on the Relative Hazard of
*
Investigations* *
*
Aspects of High-Level Radioactive Waste Disposal in
*Survey of Naturally Occurring Hazardous Materials in
*Hydrology of Some
*Disposal of Nuclear Waste by In-Situ Incorporation in
*Behavior of Candidate Canister Materials in

Cooperation and Development Countries Pursue Geological 000404
Cooperative Program on Radioactive Waste Storage in 000487
Cores from Holes P-3 HR-1 through P-3 HR-5, Area G, 000196
Cores Hanford Nuclear Reservation* 000053
Correlation of Environmental Impact with Cost of Waste 000500
Correlation of Radioactive Waste Treatment Costs and 000521
Correlation of Radioactive Waste Treatment Costs and 000652
Correlation of Radioactive Waste Treatments Costs and 000501
Correlation of Radionuclide leachabilities with 000620
Corrosion in Radioactive-Waste Storage Tanks* 000139
Corrosion of Metals in a Hot, High Pressure Brine 000130
Corrosion Considerations for Nuclear Waste Isolation 000102
Corrosion Evaluation of Stainless Steels in Idaho 000117
Corrosion Monitoring of Storage Bins for Radioactive 000116
Corrosion Resistance of Canisters for Final Disposal of 000132
Corrosion Studies Pertinent to Geologic Isolation* 000103
Corrosion Studies* 000129
Corrosion Test of a Glass-Ceramic Candidate for a 000151
Cost of Waste Treatment* 000500
Cost Estimates* 000507
Costs and the Environmental Impact of Waste Effluents 000501
Costs and the Environmental Impact of Waste Effluents 000521
Costs and the Environmental Impact of Waste Effluents 000652
Costs by Rock Type* Waste Isolation in Geologi: 000546
Costs for Radwaste Treatment* 000502
Costs* 000525
Counties, Central Kansas*1: Salt Beds in the Subsurface 000479
Counties, Mexico*of Hydrologic Conditions Near the Site 000431
Countries Pursue Geological Disposal* 000404
County, Southeastern Nevada to Investigate their 000439
County, Utah* *Subsurface Geology of a Potential 000411
Creek Clay, and Yazoo Clay in Parts of Alabama, 000430
Crib Sites* 000170
Criteria for Packaging Plutonium for Transport by Air* 000092
Criteria for Radioactive Waste Repositories* 000368
Criteria for Radioactive Waste* 000526
Criteria for Repositories for High-Level Radioactive 000435
Criteria* 000505
Criteria* *Development of 000127
Critical Masses from Stored Nuclear Waste in Natural 000015
Critical Pathways of Radionuclides to Man from Agro- 000255
Critical Study on the International Atomic Energy 000188
Criticality Monitoring Technique for a Transuranic 000650
Crust* *An Analysis of the 000414
Crystalline Rock Repository Media* *Long 000157
Crystalline Rock Using Ultrasonic Shear Waves* 000481
Crystalline Rock, Program Summary* Cooperative Program 000487
Crystalline Rocks* 000276
Crystalline Rocks: Some Geochemical, Geomechanical, and 000218
Curium in Soil-Aquatic Environments*Distribution Ratios 000301
Curium Separated from High-Level Reprocessing Waste* 000409
Curium 242 from a Contaminated Soil as a Function of pH 000288
Current High-Level Waste Solidification Technology* 000580
Current Operations of Transuranic Waste Management* 000307
Current Transuranic Waste Management on the Workers and 000209
Cursory Search for Carbonates Suitable for Developing 000371
Cycles* Contaminated Spent Fuel Reprocessing Wastes 000035
Cycles* *Estimates of Relative Areas for the Disposal 000050
Cyclone Drum Incinerator and Treatment System: July- 000622
D and D) of Retired Contaminated Facilities at Hanford* 000175
D) of Retired Contaminated Facilities at Hanford* 000175
Damage Studies on Natural and Synthetic Rock Salt 000082
Damage Studies on Natural and Synthetic Rock Salt for 000081
Decay and its Significance for Radioactive Waste Glasses 000131
December 15, 1978* Wastes Under Conditions Expected 000008
December 15, 1978* Storage Under Conditions Expected 000047
December 15, 1978-March 15, 1979* Under Conditions 000007
December 1975* Waste Management and Transportation 000226
December 1976)* for Waste Management, Production, and 000490
December 1977* *Fluidized Bed Incineration System 000349
December 1977* *Volume Reduction System for Solid and 000634
Decisions* *High-Level Nuclear 000513
Decladding Zircaloy Solid Wastes from a Fuel 000097
Decommissioning of Nuclear Facilities - An Annotated 000326
Decommissioning* 000556
Decommissioning, Long Range Plan, Idaho National 000693
Decontamination and Decommissioning, Long Range Plan, 000693
Decontamination and Melt Densification of Chop-Leach 000588
Decontamination of Alkaline Solutions of Nuclear Waste* 000486
Deep Burial of Nuclear Wastes* in Deep Geologic 000084
Deep Disposal Systems for Radioactive Wastes* 000355
Deep Exploratory Test Holes for Waste Repository 000426
Deep Geologic Alternative Actions* 000557
Deep Geologic Formations* *Environmental 000538
Deep Geologic Formations: A Perspective on the Relative 000084
Deep Mines in Precambrian Rocks* 000489
Deep Molten Silicate Rock* 000632
Deep Ocean Environments* 000188

PERMUTED TITLE INDEX

Large Nuclear-Fuel Reprocessing Plant* @Importance of
Also Geologic and Hydraulic Settings, Favorable to
for Calculating Effects of Liquid Waste Disposal in
@Nuclear Wastes in the
@Foundational Prerequisites and Practice for Using
and Its Alteration on Geological Occurrences and in
@Subsurface Disposal of Liquid Industrial Wastes by
in Deer from the Southeastern United States and in
from an @Comparison of Plutonium Concentrations in
Reservations, @Alternatives for Long-Term Management of
Processing @Alternatives for Long-Term Management of
@Summary - Alternatives for Long-Term Management of
Aiken, South @Alternatives for Long-Term Management of
@Technical Challenges in the Immobilization of Hanford
Red Incineration System for U.S. Department of Energy
Study on the International Atomic Energy Agency (IAEA)
@
of a Semi-Empirical Numerical Model for Simulating the
@
@
Waste Isolation Pilot @Experimental Studies of the
Wastes Produced in Nuclear Installations in the German
Solidified Wastes in Underground @Project Salt Vault: A
@Role of Admixtures in Preparing
@Chemical Decontamination and Melt
@Potential of
Production, and Reprocessing Division of the
@Fluidized Bed Incineration System for U.S.
on Ice - A Proposal for an International Radionuclide
@Radioactive Waste Disposal in Salt
for Underground Storage of @Evaluation of Permian Salt
Isolation of Solid Radioactive Waste in Fine-Grained
@Geologic Investigation of the Virgin River Valley Salt
@Plutonium Distribution in a Desert Pavement-
@Plutonium Distribution in a
@
Hanford* @System
@Preliminary
Resin Beads* @The
Layouts of Underground Storage Facilities* @
Repositories in Hard Rock. @Mechanical and Thermal
Formations. Volume 18. Repository Preconceptual
Formations. Volume 16. Repository Preconceptual
Formations. Volume 10. Repository Preconceptual
Geologic Formations. Volume 8. Repository Preconceptual
Formations. Volume 12. Repository Preconceptual
Heating Experiments in Hard Rock: Their Objectives and
How It Affects a Radioactive Waste Isolation Facility
@Information Base for Waste Repository
Strontium, Technetium, Cesium, Barium, @Sorption-
Waves* @Fracture
of Bulk Solids Used in Shale Fracturing Groups as
of Salt Cavity--Computer Model and Calibration from
@Cursory Search for Carbonates Suitable for
@Fire Protection
Waste Glass* @
99 Sorption on Two United States Subsoils from
Neptunium 237, and Americium 241 on Two Subsoils from
@
Cesium in @Determination of Distribution Ratios and
Fresh Water Aquifer. 1. Numerical Study of Convective-
Water Aquifer. 2. Experimental Study of Convective-
Strontium 90 and Cobalt 60 Leached from @Effect of
of Light-Water Reactor Fuel from Enriched Uranium
@Projections of Spent Fuel to be
Reprocessing Plant* @Radioactive Waste
Facilities* @Radioactive Waste
Transport Model: Theory and Numerical Implementation (
@
Radioactive Wastes* @
@Data Acquisition, Handling, and
@Ocean Waste
Laboratory* @Waste
Laboratory* @Waste
@Economic and Environmental Evaluation of Nuclear Waste
@Possibilities for Subsurface Waste
:The Research and Development Program on Waste
@Radioactive Waste
@An Appraisal of Underground Radioactive Waste
Fuel Cycles* @Estimates of Relative Areas for the
@Environmental Aspects of High-Level Radioactive Waste
@A Model for Calculating Effects of Liquid Waste
Strata* @Subsurface
@Long-Term Risk Assessment of Radioactive Waste
@The Rock-Melt Approach to Nuclear Waste
@Radioactive Waste
@Geology of Subsurface Waste
Abstracts* @Radioactive Waste

Deep Permeable Disposal Formations in Location of a 000397
Deep Placement of High-Level Radioactive Wastes* Types, 000068
Deep Saline Aquifers. Part 1 -- Development, Part 2 -- 000267
Deep Seabed* 000356
Deep Water-Bearing Horizons for Disposal of Liquid 000203
Deep-Burial Radioactive Waste Disposal* @Pollucite 000105
Deep-Well Injection* 000480
Deer from an Integrated Nuclear Fuel Cycle Facility* 000271
Deer from the Southeastern United States and in Deer 000271
Defense High-Level Radioactive Waste Hanford 000347
Defense High-Level Radioactive Waste Idaho Chemical 000346
Defense High-Level Radioactive Waste* 000478
Defense Transuranic Waste at the Savannah River Plant, 000206
Defense Waste* 000627
Defense Waste, July-December 1977* @Fluidized 000349
Definition of High-Level Radioactive Waste Unsuitable 000188
@ Deformation of a Jointed Rock Mass* 000466
Deformation Behavior of a High-Level Radioactive Waste 000020
@ Deformation Mapping of Polycrystalline Salt* 000010
@ Degradation of RAD Wastes* 000086
Degradation of RAD-Wastes for the Sandia Laboratories 000160
Democratic Republic* @Disposal of Radioactive 000050
Demonstration of the Disposal of High Activity 000316
Dense Cements for Radioactive Waste Isolation* 000662
Densification of Chop-Leach Fuel Hulls* 000588
Denver Basin for Disposal of Liquid Wastes* 000401
Department of Energy (July-December 1976)* Management, 000490
Department of Energy Defense Waste, July-December 1977* 000349
Depository in Antarctica* @Putting Radioactive Waste 000492
Deposits - A Bibliography with Abstracts* 000436
Deposits in the Texas Panhandle and Western Oklahoma 000417
Deposits* @A Concept for Long-Term 000376
Deposits, Clark County, Southeastern Nevada to 000439
Desert Bound Soil System in Area 11* 000305
Desert Pavement-Desert Bound Soil System in Area 11* 000305
Design and Fabrication of Equipment* 000531
Design for Retrieval of Solidified High-Level Wastes at 000212
Design of Experiment* 000533
Design of Systems for Handling Radioactive Ion Exchange 000670
Design Aspects of the Alpha Repository: 2. Conceptual 000581
Design Considerations for Radioactive Waste 000019
Design Studies: Basalt* Waste Isolation in Geologic 000549
Design Studies: Battelle Pacific Northwest Laboratory 000550
Design Studies: Granite* Waste Isolation in Geologic 000547
Design Studies: Salt* : Radioactive Waste Isolation in 000545
Design Studies: Shale* Waste Isolation in Geologic 000548
Design* @In Situ 000382
Design* Salt Migration to Dissolution in Salt Domes and 000263
Design, Volume 6. Excavation Technology* 000543
Desorption Studies on Granite. 1. Initial Studies of 000028
Detection in Crystalline Rock Using Ultrasonic Shear 000481
Determined by the Jenike and Johanson Method* 000641
Detroit Mine Experiment* @Growth and Shape 000462
Developing Dry Underground Storage Space in the 000371
Device for LMFBR Spent Fuel Shipping Casks* 000568
Devitrification Behavior in a Zinc Borosilicate Nuclear 000679
Differing Weathering Intensity Areas* and Technetium 000293
Differing Weathering Intensity Areas* of Technetium 99, 000294
Diffusion of Radionuclides from Geologic Formations* 000238
Diffusion Coefficients of Neptunium, Americium, and 000301
Diffusive Flow in Wellbores* and Brine Invasion into 000273
Diffusive Flow in Wellbores* Brine Invasion into Fresh 000278
Dimension of Specimen on Amounts of Cesium 137, 000639
Dioxide* "As Low As Practicable" Guides - Fabrication 000652
Discharged by the United States Nuclear Power Industry* 000690
Discharges to the Environment from a Nuclear Fuel 000633
Discharges to the Environment from Nuclear Power 000280
Discrete-Parcel-Random-Walk Version)* Mass 000215
@ Dispersion in Flow through Porous Media* 000239
@ Dispersion Relations in Injection Disposal of 000266
Display for the Master Experiments at Stripa* 000053
Disposal (A Bibliography with Abstracts)* 000367
Disposal by Shale Fracturing at Oak Ridge National 000213
Disposal by Shale Fracturing at Oak Ridge National 000218
Disposal by Underground In Situ Melting* 000498
Disposal in a Structural Syncline in Pennsylvania* 000408
Disposal in the Federal Republic of Germany* 000161
Disposal in Antarctica: Some Glaciological Aspects* 000481
Disposal in Argillaceous and Crystalline Rocks: Some 000218
Disposal in Bedded Salt of LWR Wastes from Alternative 000050
Disposal in Deep Geologic Formations* 000538
Disposal in Deep Saline Aquifers. Part 1 -- 000267
Disposal in Geologic Basins - A Study of Reservoir 000400
Disposal in Geologic Formations* 000256
Disposal in Geologic Media* 000017
Disposal in Geological Formations* 000492
Disposal in Michigan Basin* 000364
Disposal in Salt Deposits - A Bibliography with 000436

PERMUTED TITLE INDEX

of Radioactive Ions in Unsaturated Soils after Water Disposal into a Limited Surface, Part 2*Soils: Behavior 000292
Radiological Problems of Liquid Waste Disposal into the Sea 000281
Underground Disposal of Canada's Nuclear Waste 000470
States Program for the Safety Assessment of Geologic Disposal of Commercial Radioactive Wastes* *The United 000319
Underground *Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in 000316
The Disposal of High-Level Nuclear Waste in the Oceans 000477
Science Perspectives* *Geologic Disposal of High-Level Radioactive Wastes - Earth- 000362
Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste in Geologic Formations* 000268
Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste on the Ocean Floor 000261
Safety Aspect for their Exploitation for Ultimate Disposal of High-Level Waste* Salt Domes, An Important 000407
North Flank of Appalachian Basin in *Possibilities for Disposal of Industrial Wastes in Subsurface Rocks on 000429
The Disposal of Iodine 129 000062
Problems in the Disposal of Irradiated Fuel Cladding 000416
Injection* *Subsurface Disposal of Liquid Industrial Wastes by Deep-Well 000280
and Practice for Using Deep Water-Bearing Horizons for Disposal of Liquid Radioactive Wastes* Prerequisites 000203
Potential of Denver Basin for Disposal of Liquid Wastes 000401
* Disposal of Long-Lived Radioactive Wastes in Italy* 000403
Facilities and General *Engineered Storage for the Disposal of Non-Fuel Bearing Components from Nuclear 000098
in Volume Reduction, Packaging, Transportation, and Disposal of Non-Fuel Bearing Hardware Resulting from 000651
Deep Molten Silicate Rock* * Disposal of Nuclear Waste by In-Situ Incorporation in 000632
Sea Disposal of Packaged Radioactive Wastes 000518
Summary* *Feasibility of Space Disposal of Radioactive Nuclear Waste. 1. Executive 000428
Summary* *Feasibility of Space Disposal of Radioactive Nuclear Waste. 2. Technical 000433
The Disposal of Radioactive Waste on Land 000471
the Appalachian Basin with Reference to the Subsurface Disposal of Radioactive Waste Solutions* Summary of 000379
Earth Science Technical Plan for Mined Geologic Disposal of Radioactive Waste 000503
Oceanic Research Needed for Safe Disposal of Radioactive Wastes at Sea 000449
Recent Results and Developments on the Disposal of Radioactive Wastes in the Asse Salt Mine 000177
* Disposal of Radioactive Wastes in Salt Domes* 000024
Evaluation on the Disposal of Radioactive Wastes into the North Pacific 000303
Installations in the German Democratic Republic* * Disposal of Radioactive Wastes Produced in Nuclear 000450
Dispersion Relations in Injection Disposal of Radioactive Wastes 000266
The Brookhaven Experience with Sea Disposal of Radioactive Wastes 000598
the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes* *An Investigation of 000428
Republic of Germany. Vol 4. The Law Concerning t Disposal of Radioactive Wastes* Wastes in the Federal 000495
The Canadian Program for Storage a Disposal of Spent Fuel and High-Level Wastes 000329
Corrosion Resistance of Canisters for Final Disposal of Spent Nuclear Fuel 000132
Alternatives Involving Retrieval, Processing, and Disposal on the Idaho National Engineering Laboratory 000342
A Review of Radioactive Waste Disposal to the Ground at Hanford 000191
Studies on Natural and Synthetic Rock Salt for Waste Disposal Applications* *Radiation Damage 000081
Allows Objective *Comparison of Some Geologic and Ocean Disposal Concepts Regarding Realistic Modeling that 000517
Nuclear Shipping and Waste Disposal Cost Estimates 000507
of Actinide-Bearing Sediments Underlying Liquid Waste Disposal Facilities at Hanford* *Characterization 000290
United States Seabed High-Level Radioactive Waste Disposal Feasibility Study 000304
Reprocessing Plant* *Importance of Deep Permeable Disposal Formations in Location of a Large Nuclear-Fuel 000391
Interim Report* *Hydrology of Waste Disposal National Reactor Testing Station, Idaho, An 000418
Oceanographic Research on Waste Disposal Off the Coast of Tokai-Mura 000415
Canada Keeps its Disposal Options Open 000473
Modelling Studies Used to Evaluate Waste Disposal Options 000260
Subsurface Waste- Disposal Potential in Saline Basin of Kansas 000394
of Kansas Evaporites - The Radioactive Waste Disposal Problem* *Dissolution 000459
Waste Disposal Problems 000331
Estimate Magnitude of Waste Disposal Problems 000698
Column Experiment* *Radioactive Waste Disposal Report on Seepage Pit Liquid Waste-Shale 000178
Waste* *Antarctica, A Potential Disposal Site for the World's High-Level Radioactive 000491
at Commercially Operated Low-Level Radioactive Waste Disposal Sites* Migration-Land Burial Water Chemistry 000166
at Commercially Operated Low-Level Radioactive Waste Disposal Sites* Migration-Land Burial Water Chemistry 000168
Chemistry at Commercially Low-Level Radioactive Waste Disposal Sites* of Isotope Migration-Land Burial Water 000167
at the New Hungarian Radioactive Waste Handling and Disposal Station* *Technological Facilities 000593
Underground Waste Disposal Studies, Chemical Processing Plant Area 000192
Generating Plant Liquid and Solid Radioactive Waste Disposal Systems* *Operation of the Monticello Nuclear 000595
Deep Disposal Systems for Radioactive Wastes 000355
High-Level and Long-Lived Radioactive Waste Disposal 000351
Salt Motion Following Nuclear Waste Disposal 000386
A Bibliography on Ocean Waste Disposal 000865
Review of Rock Mechanics Workshop on Radioactive Waste Disposal *Summary 000373
of High-Level Atomic Wastes with the View to Ultimate Disposal* *Processing 000609
Alteration: An Implication for Radioactive Waste Disposal* *Metamict Mineral 000030
Waste Monazites, Ideal Actinide Hosts for Geologic Disposal* *Synthesis of Nuclear 000680
Cooperation and Development Countries Pursue Geological Disposal* *Organization for Economic 000808
Aluminosilicate Adsorbent for Radioactive Waste Disposal* *Development of an Effective 000606
Occurrences and in Deep-Burial Radioactive Waste Disposal* *Pollucite and Its Alteration on Geological 000105
*Are Plutons the Answer to Nuclear Waste Disposal? 000318
Nuclear Waste Disposal: The Geological Aspects 000375
Concerning the Possibility of Disposing of Radioactive Waste in Ocean Trenches 000357
Facilities at Hanford* * Disposition (D and D) of Retired Contaminated 000175
Kansas, Part 1: Salt Beds in the Subsurface in *Salt Dissolution in Oil and Gas Test Holes in Central 000479
Salt Dissolution in Oil and Gas Test Holes in Kansas 000478
Radioactive Waste *The Response of Salt Migration to Dissolution in Salt Domes and How It Effects a 000263
Waste Disposal Problem* * Dissolution of Kansas Evaporites - The Radioactive 000459
National Waste Terminal Storage Review Meeting on Salt Dissolution 000348
Cannikin Nuclear *Hydrologic Processes and Radionuclide Distribution in a Cavity and Chimney Produced by the 000377
System in Area 11* *Plutonium Distribution in a Desert Pavement-Desert Round Soil 000305
Environment* *The Distribution of Plutonium in a Rock Containment 000031
Testing* *Oceanic Distribution of Radionuclides from Nuclear Weapons 000230
Neptunium, Americium, and Curium in *Determination of Distribution Ratios and Diffusion Coefficients of 000301
for Waste Management, Production, and Reprocessing Division of the Department of Energy (July-December 000490
March 1978* *Chemical Engineering Division Fuel Cycle Program Progress Report, January- 000336

PERMUTED TITLE INDEX

at the Idaho National Engineering *Safety Review
Deep Saline Aquifers. Part 1 -- Development, Part 2 --
of the Interior Salt Domes of Northeast Texas Salt
*Geology of Certain Domes of the Northeast Texas Salt-
*Regional Geology of the Northeast Texas Salt
and Growth of Salt Domes in the Northeast Texas Salt
*The Response of Salt Migration to Dissolution in Salt
*An Investigation of the Utility of Gulf Coast Salt
*Genesis and Growth of Salt
*Geology of Certain
Their Suitability *Geologic Study of the Interior Salt
*Disposal of Radioactive Wastes in Salt
*Thermal Conductivity of Rock Salt from Louisiana Salt
*Geohydrologic Studies of Gulf Coast Interior Salt
Exploitation for *The Waterproof Geometry of Salt
Receiving Area* *Radiation
*Statement of
*Statement of
Commercial Waste Management: *Contribution to
Concerning the Evolution in the Ground of *Conclusions
of Fission Product Solutions Using a Drum
Air Cooling of a Waste Isolation Pilot Plant Mine
*Petrology of Tuff Units from the J-13
*Borehole
the Solidification of Fission Product Solutions Using a
1978* *TRU Waste Cyclone
*Initial
in Waste* *An Automatic
*Cursory Search for Carbonates Suitable for Developing
of High-Level Radioactive Waste Unsuitable for
Radioactive Wastes* *
*Annual Report on
*Statement of Dr. James
*
Disposal of Radioactive Waste* *
*Geologic Disposal of High-Level Radioactive Wastes -
of the Measured Values for the State of Stress in the
Feasibility of Sealing Boreholes with Compacted Natural
Feasibility of Sealing Boreholes with Compacted Natural
Bodies from the Appalachian Mountain Region of
Disposal by Underground In Situ Melting* *
and Reservoir Strata* *
Radioactive Wastes in *A Preliminary Engineering and
Geological Disposal* *Organization for
Colorado* *Sedimentary History and
of Low-level Radionuclide Solutions into a Stream
*Critical Pathways of Radionuclides to Man from Agro-
Near the Site of a Potential Nuclear Waste Repository,
Waste Disposal* *Development of an
in Geologic Formations. Volume 23 - Environmental
*
*Summary of Radioactivity Released in
Treatment Costs and the Environmental Impact of Waste
Treatments Costs and the Environmental Impact of Waste
Treatment Costs and the Environmental Impact of Waste
Geologic Media*Electron Transfer Method of Controlling
of Stress-Strain Data Suitable for Finite-Element
the Transient Method* *Measuring the Permeability of
*Studies of the
Study of the Terminal Waste Storage Program,
Stripa Mine* *
Synthetic Rock Salt Utilizing Measurements Made During
Adsorption of Multivalent Elements by Geologic Media* *
*Experimental Permeability Studies at
of Physical Chemistry of Fluids in Rock Salt at
*Radioactive Waste Fixation in FUETAP (Formed Under
*Long-Term
in Salt Beds in the Subsurface in Russell, Lincoln,
High-Level Waste Glasses by Sputter-Induced Optical
Analysis of the Back End of the Nuclear Fuel Cycle with
Deformation *Interim Report on Development of a Semi-
*High-Level Waste Canister
and Tennessee as Potential Host Rocks for Underground
Utah* *Subsurface Geology of a Potential Waste
*Metal
and Reprocessing Division of the Department of
*Wet
*Storage and Release of Radiation
*A Critical Study on the International Atomic
of the Hanford Reservation of the U.S. Atomic
Bed Incineration System for U.S. Department of
Technological Program for *Description of the
*Environmental Geology in Development of Nuclear
Components from Nuclear Facilities and General *
High-Level Radioactive Wastes in Concrete*A Preliminary
Report, January-March 1978* *Chemical
Lesbi Range)**Present Environment of the Idaho National
*Identification of Alternatives for Idaho National

Document for the Radioactive Waste Management Complex 000176
Documentation* Effects of Liquid Waste Disposal in 000267
Dome Basin to Investigate Their Suitability for 000039
Dome Basin* 000352
Dome Basin* 000412
Dome Basin* *Genesis 000432
Domes and How It Affects a Radioactive Waste Isolation 000263
Domes for the Storage or Disposal of Radioactive Wastes* 000428
Domes in the Northeast Texas Salt Dome Basin* 000432
Domes of the Northeast Texas Salt-Dome Basin* 000353
Domes of Northeast Texas Salt Dome Basin to Investigate 000438
Domes* 000038
Domes* 000061
Domes* 000460
Domes, An Important Safety Aspect for their 000407
Domes to Rodents Inhabiting a Radioactive Waste 000283
Dr. James E. Martin* 000518
Dr. Terry R. Lash* 000511
Draft Generic Environmental Impact Statement on 000457
Drawn from Studies on the Oklo Natural Nuclear Reactor 000287
Drier* *IPS - A Process for the Solidification 000607
Drift* *Forced- 000522
Drill Site, Jackass Flats, Nevada* 000410
Drilling and Related Activities at the Stripa Mine* 000422
Drum Drier* *PIPS - A Process for 000607
Drum Incinerator and Treatment System: July-September, 000422
Drum Retrieval Final Report* 000185
Drum Scanning System for the Measurement of Plutonium 000106
Dry Underground Storage Space in the Midcontinent, U.S. 000371
Dumping at Sea* Atomic Energy Agency (IAEA) Definition 000188
Durability of Containers for Storing Sol Jified 000095
DWMT PROJECT AN0115A, FY-1975* 000252
E. Martin* 000518
Early Waste Retrieval Interim Report* 000186
Earth Science Technical Plan for Mined Geologic 000507
Earth-Science Perspectives* 000362
Earth's Crust* *An Analysis 000318
Earthen Material, Volumes 1 and 2* * 000352
Earthen Materials, Volume 3* * 000366
Eastern United States with Regard to Siting of 000388
Economic and Environmental Evaluation of Nuclear Waste 000399
Economic and Industrial Potential of Geologic Basins 000399
Economic Analysis of the Fixation of High-Level 000685
Economic Cooperation and Development Countries Pursue 000404
Economic Geology of San Juan Basin, New Mexico and 000383
Ecosystem* *A Transport Model for Long Term Release 000311
Ecosystems* 000255
Eddy and Lea Counties, Mexico* of Hydrologic Conditions 000331
Effective Aluminosilicate Adsorbent for Radioactive 000606
Effluent Analyses* Radioactive Waste Isolation 000296
Effluents from the Waste Isolation Facility* 000558
Effluents from Nuclear Power Plants from 1973 thru 1976* 000708
Effluents in the Nuclear Fuel Cycle - Conversion of 000521
Effluents in the Nuclear Fuel Cycle - Reprocessing of 000501
Effluents in the Nuclear Fuel Cycle for Use in 000652
Eh During Adsorption of Multivalent Elements by 000073
Elastic-Plastic Analysis of Shipping Containers* 000183
Eleana Argillite from Area 17, Nevada Test Site, Using 000088
Eleana Argillite* 000423
Eleana Investigation* *Linearity 000396
Electrical Heaters for Thermo-Mechanical Tests at the 000538
Electron Irradiation* Damage Studies on Natural and 000082
Electron Transfer Method of Controlling Eh During 000076
Elevated Temperature and Pressure of Granitic Rocks* 000069
Elevated Temperature and Pressure to Repositories for 000080
Elevated Temperature and Pressure) Concretes - 000060
Elevated Temperature Leaching of Solid Waste Forms* 000124
Ellsworth, Barton, and Rice Counties, Central Kansas* 000479
Emission* of Leached Surface Layers on Simulated 000118
Emphasis on High-Level Waste Management. Volume 2* *An 000321
Empirical Numerical Model for Simulating the 000020
Emplacement and Retrieval Concepts Study* 000561
Emplacement of Waste* in Parts of Alabama, Mississippi 000430
Emplacement Site, Salt Valley Anticline, Grand County, 000411
Encapsulation of Ceramic Nuclear* 000618
Energy (July-December 1976)* Management, Production, 000490
Energy from Nuclear Power* 000520
Energy in Salt in Radioactive Waste Repositories* 000042
Energy Agency (IAEA) Definition of High-Level 000188
Energy Commission, Washington* Water Characteristics 000189
Energy Defense Waste, July-December 1977* *Fluidized 000283
Energy Research and Development Administration 000325
Energy* 000452
Engineered Storage for the Disposal of Non-Fuel Bearing 000098
Engineering and Economic Analysis of the Fixation of 000685
Engineering Division Fuel Cycle Programs Progress 000336
Engineering Laboratory (Including RWMC, Site 14, and 000207
Engineering Laboratory Transuranic Waste Management* 000475

for Long-Term Management of Stored Idaho National
of Buried Transuranic Waste at the Idaho National
and Decommissioning, Long Range Plan, Idaho National
Processing, and Disposal on the Idaho National
Waste Management Complex at the Idaho National
Level Transuranic Waste at the Idaho National
Waste Depository* *Computer
Guides - Fabrication of Light-Water Reactor Fuel from
*Radioactive Waste Discharges to the
*Radioactive Waste Discharges to the
Laboratory (Including RWMC, Site 14, and Lemhi *Present
*The Distribution of Plutonium in a Rock Containment
*Corrosion of Metals in a Hot, High Pressure Brine
*Nuclear Safety and the
Long-Term Management of Stored Idaho National *
Disposal in Deep Geologic Formations* *
Waste Burial* *
Transuranic Waste Management* *
Transuranic Waste* *
Waste Isolation in Geologic Formations. Volume 23 -
Underground In Situ Melting* *Economic and
*Analysis of Plutonium-Release Consequences on the
*
of Radioactive Waste Treatment Costs and the
of Radioactive Waste Treatment Costs and the
of Radioactive Waste Treatment Costs and the
*Correlation of
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste Isolation in *Technical Support for Generic
Waste in Geologic *Technical Support for Generic
Management: Radioactive *Contribution to Draft Generic
Isolation in Geologic *Technical Support for Generic
Isolation in Geologic *Technical Support for Generic
Isolation in Geologic *Technical Support for Generic
*
*Methodology for the Determination of
Plant Site, Calendar Year 1976* *
*
Reprocessing and High Level Waste *Description of the
*Considerations of
Wastes* *
Management* *An Assessment Methodology of
*Determination of Transuranics in
*Behavior of Candidate Canister Materials in Deep Ocean
of Neptunium, Americium, and Curium in Soil-Aquatic
*Montmorillonite Exchange
*The Fundamentals of Ion Exchange
*Linde SIX
*Design and Fabrication of
*Installation of Experimental
*
Isolation Pilot *Batch Kd Measurements of Nuclides to
*Hanford Groundwater Transport
Plastic Analysis of Waste-Container Sleeve: 1. Initial
Salt of LWR Wastes from Alternative Fuel Cycles* *
*Nuclear Shipping and Waste Disposal Cost
Waste Management* *Methodology for
Migration in Aerated Zones. 2. Theoretical
*Techniques for High-Level Waste Solidification in
*Radioactive Waste Management Development in
of Strontium, Technetium, Cesium, Barium, Cerium,
*Modelling Studies Used to
Stored Idaho National *Environmental and Other
*Liquid Waste
Radioactive Waste* *Putting
Wastes Using Mechanically Agitated Thin-Film
*Dissolution of Kansas
on the Oklo Natural Nuclear Reactor Concerning the
Information Base for Waste Repository Design, Volume 6.
of Space Disposal of Radioactive Nuclear Waste. 1.
: Radioactive Waste in Geologic Formations. Volume 1.
*Description of Solid Waste
Microbial Impact on Transuranic Wastes Under Conditions
Microbial Impact on Transuranic Wastes Under Conditions
Microbial Impact on Transuranic Wastes Under Conditions
*Studies of Transuranic Waste Storage Under Conditions
Wastes at the Nuclear Research Center Karlsruhe -
Control at the Stage of Transporting Solid Radioactive *

Engineering Laboratory Transuranic Waste* Alternatives 000380
Engineering Laboratory* *Retrieval 000184
Engineering Laboratory* *Decontamination 000693
Engineering Laboratory* Involving Retrieval, 000382
Engineering Laboratory* Document for the Radioactive 000176
Engineering Laboratory* Interim Storage of Intermediate- 000187
Enhanced "Release Scenario" Analysis for a Nuclear 000560
Enriched Uranium Dioxide* "As Low As Practicable" 000652
Environment from a Nuclear Fuel Reprocessing Plant* 000633
Environment from Nuclear Power Facilities* 000280
Environment of the Idaho National Engineering 000207
Environment* 000031
Environment* 000130
Environment* 000332
Environmental and Other Evaluations of Alternatives for 000380
Environmental Aspects of High-Level Radioactive Waste 000538
Environmental Assessment Related to Hanford Radioactive 000268
Environmental Effects from Current Operations of 000307
Environmental Effects of Alternatives for Managing 000308
Environmental Effluent Analyses* Statement: Radioactive 000296
Environmental Evaluation of Nuclear Waste Disposal by 000098
Environmental Geochemistry* 000270
Environmental Geology in Development of Nuclear Energy* 000452
Environmental Impact of Waste Effluents in the Nuclear 000501
Environmental Impact of Waste Effluents in the Nuclear 000521
Environmental Impact of Waste Effluents in the Nuclear 000652
Environmental Impact with Cost of Waste Treatment* 000500
Environmental Impact Statement (GEIS): Radioactive 000021
Environmental Impact Statement (GEIS): Radioactive 000022
Environmental Impact Statement (GEIS): Radioactive 000023
Environmental Impact Statement (GEIS): Radioactive 000024
Environmental Impact Statement (GEIS): Radioactive 000237
Environmental Impact Statement (GEIS): Radioactive 000380
Environmental Impact Statement (GEIS): Radioactive 000385
Environmental Impact Statement (GEIS): Radioactive 000585
Environmental Impact Statement (GEIS): Radioactive 000586
Environmental Impact Statement (GEIS): Radioactive 000587
Environmental Impact Statement (GEIS): Radioactive 000588
Environmental Impact Statement (GEIS): Radioactive 000589
Environmental Impact Statement (GEIS): Radioactive 000552
Environmental Impact Statement on Commercial Waste 000857
Environmental Impact Statement: Radioactive Waste 000075
Environmental Impact Statement: Radioactive Waste 000296
Environmental Impact Statement: Radioactive Waste 000550
Environmental Impact* 000238
Environmental Iodine 129 and Technetium 99* 000008
Environmental Monitoring at the Waste Isolation Pilot 000363
Environmental Monitoring* 000555
Environmental Protection Agency Program for 000324
Environmental Protection Criteria for Radioactive Waste* 000526
Environmental Protection Requirements for Radioactive 000515
Environmental Risks Associated with Radioactive Waste 000259
Environmental Water* 000600
Environments* 000188
Environments* Ratios and Diffusion Coefficients 000307
Equilibria with Strontium-Sodium-Cesium* 000027
Equilibria* 000676
Equilibrium with Trivalent Cerium and Yttrium Cations* 000002
Equipment* 000531
Equipment* 000539
Estimate Magnitude of Waste Disposal Problem* 000698
Estimate Migration Potential at the Proposed Waste 000300
Estimates of Hypothetical Radioactive Waste Incidents* 000221
Estimates of Loading on the Sleeve* *Thermoelastic/ 000067
Estimates of Relative Areas for the Disposal in Bedded 000050
Estimates* 000307
Estimating Accidental Radioactive Releases in Nuclear 000278
Estimation of Strontium 90 Migration* *Radionuclides 000288
Europe* 000611
Europe* 000688
Europium, Uranium, Plutonium, and Americium* Studies 000028
Evaluate Waste Disposal Options* 000260
Evaluations of Alternatives for Long-Term Management of 000380
Evaporation and Acid Recovery* 000666
Evaporators to Work: Concentration of Aqueous 000603
Evaporators* *Volume Reduction of Liquid Radioactive 000592
Evaporites - The Radioactive Waste Disposal Problem* 000459
Evolution in the Ground of Radioactive Waste over a 000287
Excavation Technology* * 000583
Executive Summary* *Possibility 000038
Executive Summary* Environmental Impact Statement (GEIS) 000552
Expected from a Large Commercial Reprocessing Plant* 000762
Expected in the Waste Isolation Pilot Plant (WIPP). 000007
Expected in the Waste Isolation Pilot Plant (WIPP). 000008
Expected in the Waste Isolation Pilot Plant (WIPP). 000009
Expected in the Waste Isolation Pilot Plant (WIPP). 000047
Experience from Plant Operation and Development Work* 000612
Experience in the Organization of Sanitary Radiometric 000567

PERMUTED TITLE INDEX

Transportation, and Disposal of Non-Fuel Bearing 0
0Relevance of Natural Retention
Vitrification of High-Level 0Non-Radioactive Operation
0The Brookhaven
Radioactive Wastes in 0The Scientific Basis for, and
0Operating
0Operating Practices and
Testing Station* 0Operating Practices,
of Salt Dome, Is Important Safety Aspect for their
Investigations* 0Deep
a Cavity and Chimney Produced by the Cannikin Nuclear
Contaminated Soil as a Function of pH and Certain Soil 0
0Design and
Use in Establishing "As Low As Practicable" Guides -
0Waste Incineration and Immobilization for Nuclear
0Decommissioning of Nuclear
Disposal of Non-Fuel Bearing Components from Nuclear
Handling and Disposal Station* 0Technological
0Disposition (D and D) of Retired Contaminated
Bearing Sediments Underlying Liquid Waste Disposal
0Above-Ground Storage
0Description of Waste Isolation
0Contaminated Reactor
Handling and Packaging Program. A Survey of Hot Cell
Waste Discharges to the Environment from Nuclear Power
and Inactive Mines for Possible Use as In Situ Test
2. Conceptual Layouts of Underground Storage
0Reprocessing
Retrieved from the Second Calcined Solids Storage
: Radioactive Waste Isolation in Geologic Formations.
Areas and How It Affects a Radioactive Waste Isolation
0A Continuous Tritium Monitor for Nuclear
in Nuclear Fuel Reprocessing and Waste Solidification
0Function and Licensing Status of the Asse Test Storage
0Effluents from the Waste Isolation
Safety Assessment of the Waste Isolation Pilot Plant
and in Beer from an Integrated Nuclear Fuel Cycle
0Potential Mechanisms of Containment
Waste Idaho Chemical Processing Plant, Idaho
0Geology of the 241-AX Tank
0Geology of the 241-B Tank
0Geology of the 241-BX Tank
Plants 0The Relationship of Microbial Processes to the
0Nuclide Migration Models for
0Rock Types, Also Geologic and Hydraulic Settings,
Isolation in Geologic Formations. Facility Construction
Asphalt or Polyethylene* 0The
Earthen Material, Volumes 1 and 2* 0
Earthen Materials, Volume 3* 0
Waste. 2. Technical Summary* 0
Waste. 1. Executive Summary* 0
River Basalts* 0The
Appalachian Mountain 0Supplementary Report: Geologic
Blocks Containing Radioactive Wastes* 0
States Seabed High-Level Radioactive Waste Disposal
0Basalt Waste Isolation Program Monthly Report,
Repositories* 0State/
Involving Retrieval, Processing, and Shipment to the
the Disposal 0System Study, Radioactive Wastes in the
and Development Program on Waste Disposal in the
of Germany: Accumulation of Radioactive Wastes in the
Radioactive Wastes in the 0Radioactive Wastes in the
Experiment* 0
0Theoretical Temperature
Radioactive Glass Blocks Buried Below the Water Table:
Radioactive Wastes Using Mechanically Agitated Thin-
0Corrosion Resistance of Canisters for
Cesium Separated from High-Level Reprocessing Waste* 0
0Western New York Nuclear Service Center Study,
0Initial Drum Retrieval
Waste. B. Safety 0Handling of Spent Nuclear Fuel and
0Considerations Relevant to Provisional versus
for Long-Term Isolation of Solid Radioactive Waste in
0
0An Assessment of Stress-Strain Data Suitable for
0Material Transport through Porous Media: A
Product Solutions Using a Drum Drier* 0
Casks* 0
0Studies on Fixation of Radioactivity in
from High-Level Waste During Transportation
0Waste-Isolation Projects:
0FIPS - A Process for the Solidification of
0Accumulation and Movement of
0The Behavior of Radioactive
September 30, 1979* 0Oklo -- Natural
0Analysis of Some Tracer Data in Granite Rock Using a
Reprocessing 0Status Report, Waste Incineration and
0Evaluation of Cesium 137 Sorbents for

Experience in Volume Reduction, Packaging, 000651
Experience to Nuclear Waste Management* 000548
Experience with a Joule Heated Ceramic Melter for 000686
Experience with Sea Disposal of Radioactive Wastes* 000598
Experience With Underground Storage of Liquid 000068
Experience* 000189
Experiences at Hanford* 000179
Experiences, and Problems at the National Reactor 000200
Exploitation for Ultimate Disposal of High-Level Waste* 000007
Exploratory Test Holes for Waste Repository 000426
Explosion, Amchitka Island, Alaska* Distribution in 000377
Extractability of Plutonium 238 and Curium 242 from a 000200
Fabrication of Equipment* 000531
Fabrication of Light-Water Reactor Fuel from Enriched 000652
Facilities (April-September 1977)* 000616
Facilities - An Annotated Bibliography* 000326
Facilities and General Radioactive Waste* for the 000094
Facilities at the New Hampshire Radioactive Waste 000593
Facilities at Hanford* 000175
Facilities at Hanford* 0Characterization of Actinide- 000290
Facilities for Vitrified Highly Active Wastes* 000573
Facilities* 000553
Facilities* 000692
Facilities* 0Spent Fuel 000135
Facilities* 0Radioactive 000200
Facilities* 0Survey of Active 000378
Facilities* 0Design Aspects of the Alpha Repository: 000541
Facilities, Idaho Chemical Processing Plant* 000503
Facility at Idaho Chemical Processing Plant* Calcine 000200
Facility Construction Feasibility and Costs by Rock Type 000586
Facility Design* Salt Migration to Dissolution in Salt 000163
Facility Off-Gas Streams* 000172
Facility Off-Gas* of Technetium 99 and Selenium 79 000594
Facility* 000897
Facility* 000558
Facility* 0Preliminary 000094
Facility* in Deer from the Southeastern United States 000271
Failure* 000558
Falls, Idaho* of Defense High-Level Radioactive 000346
Farm* 000193
Farm* 000194
Farm* 000195
Fate and Behavior of Transuranic Elements in Soils, 000310
Faulted Media* 000232
Favorable to Deep Placement of High-Level Radioactive 000068
Feasibility and Costs by Rock Type* : Radioactive Waste 000586
Feasibility of Incorporating Radioactive Wastes in 000596
Feasibility of Sealing Boreholes with Compacted Natural 000052
Feasibility of Sealing Boreholes with Compacted Natural 000066
Feasibility of Space Disposal of Radioactive Nuclear 000433
Feasibility of Space Disposal of Radioactive Nuclear 000434
Feasibility of Storing Radioactive Wastes in Columbia 000406
Feasibility of Talc and Serpentinite Bodies from the 000484
Feasibility Study of Underground Storage of Bitumen 000581
Feasibility Study* 0United 000304
February 1978* 000387
Federal Licensing Interactions for Radioactive Waste 000504
Federal Repository* 0Alternatives 000341
Federal Republic of Germany. Vol 4. The Law Concerning 000495
Federal Republic of Germany* 0The Research 000161
Federal Republic of Germany* in the Federal Republic 000696
Federal Republic of Germany: Accumulation of 000696
Field and Laboratory Studies Preliminary to Main 000012
Fields for the Strips Waster Project, Volume 1* 000374
Fifteen Years of Results* of Radioactivity from Highly 000685
Film Evaporators* 0Volume Reduction of Liquid 000592
Final Disposal of Spent Nuclear Fuel* 000132
Final Purification and Concentration of Americium/ 000489
Final Report for Public Comment* 000211
Final Report* 000185
Final Storage of Vitrified High-Level Reprocessing 000123
Final Storage* 000559
Fine-Grained Deposits* 0A Concept 000376
Fine-Grained Sheet Silicate Rocks* 000483
Finite-Element Elastic-Plastic Analysis of Shipping 000183
Finite-Element Galerkin Model* 000283
FIPS - A Process for the Solidification of Fission 000607
Fire Protection Device for LWBR Spent Fuel Shipping 000560
Fired Clays* 000678
Fires* 0Release of Radionuclides 000110
Fiscal Year 1978* 000487
Fission Product Solutions Using a Drum Drier* 000607
Fission Products in Soils and Plants* 000058
Fission Products in Soils* 000272
Fission Reactor Program, Progress Report for July 1- 000065
Fissure Model* 000287
Fixation for Waste Management, Production, and 000490
Fixation in Concrete* 000655

PERMUTED TITLE INDEX

and Pressure) Concretes -	*Radioactive Waste	Fixation in PUETAP (Formed Under Elevated Temperature	000060
of Medium Level Wastes and Spent Solvent and on		Fixation into Bitumen* Experiments on the Treatment	000578
8) Preliminary Engineering and Economic Analysis of the		Fixation of High-Level Radioactive Wastes in Concrete*	000685
	*Sorption and	Fixation of Large Cations by Shale Formations*	000014
with Clays*	*	Fixation of Radioactive Waste by Hydrothermal Reactions	000575
	*The	Fixation of Radioactive Waste for Posterity*	000485
Experimental Evaluation of Phosphate and Porosilicate	*	Fixation of Radioactive Waste in Glass. Part 2. The	000108
	*	Fixation of Radioactive Wastes in Clay-Flux Mixes*	000642
	*Heat Experiments for Self	Fixation of Radioactive Wastes in Clay-Flux Mixtures*	000617
	*Studies on	Fixation of Radioactivity in Fired Clays*	000670
	*	Fixation of Radioactivity in Stable, Solid Media*	000646
	*	Fixation of Radioactivity in Stable, Solid Media*	000680
	*	Fixation of Radioisotopes on Hanford Soil Types*	000183
of Industrial Wastes in Subsurface Rocks on North		Flank of Appalachian Basin in New York* for Disposal	000429
*Radioactive Waste Package Development at the Rocky		Flats Plant*	000136
of Tuff Units from the J-13 Drill Site, Jackass		Flats, Nevada* *Petrology	000410
Prototype Sites of Repositories for Radioactive Wastes:		Flood Basalt* *Hypothetical	000444
Aspects of Disposal of High-Level Waste on the Ocean		Floor* *Assessment of the Radiological Protection	000261
	*	Flow in Fractured Porous Media*	000282
Aquifer. 1. Numerical Study of Convective-Diffusive		Flow in Wellbores* and Brine Invasion into Fresh Water	000273
Aquifer. 2. Experimental Study of Convective-Diffusive		Flow in Wellbores* and Brine Invasion into Fresh Water	000274
	*Dispersion in	Flow through Porous Media*	000239
	*Compressible Fluid	Flow Through Rocks of Variable Permeability*	000049
Tracts as Determined by the Jenike and Johanson Method*	*	Floatability of Bulk Solids Used in Shale Fracturing	000641
Parts of *Basal Ottawa Limestone, Chattanooga Shale,		Floyd Shale, Porters Creek Clay, and Yazoo Clay in	000438
*Trace-Element Transport in Lithic Material by		Fluid at High Temperatures*	000297
	*Compressible	Fluid Flow Through Rocks of Variable Permeability*	000049
Problems of Siting the Waste *Application of Studies of		Fluid Inclusions in Permian Salado Salt, New Mexico, to	000372
of Energy Defense Waste, July-December 1977*	*	Fluidized Bed Incineration System for U.S. Department	000349
Pressure to *Application of Physical Chemistry of		Fluids in Rock Salt at Elevated Temperature and	000080
of Yellow Cake to Uranium Hexafluoride. Part 1. The		Fluorination-Fractionation Process* Cycle - Conversion	000521
*Fixation of Radioactive Wastes in Clay-		Flux Mixes*	000642
for Self Fixation of Radioactive Wastes in Clay-		Flux Mixtures* *Heat Experiments	000617
*Polonium and Plutonium in An Intertidal		Food Chain*	000235
Mine Drift*	*	Forced-Air Cooling of a Waste Isolation Pilot Plant	000532
Zirconia Wastes* *Development of a Pelleted Waste		Form for High-Level Idaho Chemical Processing Plant	000628
of Radionuclide Migration in a Homogeneous Clayey		Formation (Application to the Case of the Boom Clay at	000291
Radioactive Waste Glasses* *Helium		Formation from Alpha-Decay and its Significance for	000131
*A Waste Repository in a Geologic		Formation*	000529
on the System Liquid Radioactive Wastes - Geologic		Formation* *Influence of Radiation	000078
Radioactive Waste Repository in a Bedded Salt		Formation*of the Geological Stability of a Hypothetical	000562
*Waste Isolation in Geologic		Formations in the United States of America*	000528
Reprocessing *Importance of Deep Permeable Disposal		Formations in Location of a Large Nuclear-Fuel	000391
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Baseline Rock Properties--Granite*Statement	000021
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Facility Construction Feasibility and Costs	000546
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Stratigraphies of Salt, Granite, Shale, and	000385
Impact Statement (GEIS): Radioactive Waste in Geologic		Formations. Volume 1. Executive Summary* Environmental	000552
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 10. Repository Preconceptual Design	000547
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 12. Repository Preconceptual Design	000548
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 18. Repository Preconceptual Design	000549
Statement: Radioactive Waste Isolation in Geologic		Formations. Volume 16. Repository Preconceptual Design	000550
Statement: Radioactive Waste Isolation in Geologic		Formations. Volume 19 - Thermal Analyses* Impact	000075
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 20. Thermomechanical Stress Analysis	000024
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 21. Ground Water Movement and	000237
Statement: Radioactive Waste Isolation in Geologic		Formations. Volume 23 - Environmental Effluent Analyses*	000296
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 4. Baseline Rock Properties: Salt*	000384
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 6. Baseline Rock Properties - Shale*	000022
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 7. Baseline Rock Properties - Basalt*	000023
(GEIS): Radioactive Waste Isolation in Geologic		Formations. Volume 8. Repository Preconceptual Design	000545
*Sorption and Fixation of Large Cations by Shale		Formations*	000014
*Light-Water Reactor Wastes to be Isolated in Geologic		Formations*	000786
*Diffusion of Radionuclides from Geologic		Formations*	000234
*Radioactive Waste Disposal in Geological		Formations*	000402
*Terminal Storage of Radioactive Wastes in Geologic		Formations*	000425
Storage of High-Level Waste Solids and Liquids in Salt		Formations* *Ultimate	000359
Storage of High-Level Waste Solids and Liquids in Salt		Formations* *Ultimate	000360
Assessment of Radioactive Waste Disposal in Geologic		Formations* *Long-Term Risk	000256
High-Level Radioactive Waste Disposal in Deep Geologic		Formations* *Environmental Aspects of	000538
Solid High-Level and Alpha-Bearing Wastes in Geological		Formations* *Site Selection for Repositories of	000413
Management: Radioactive Waste Isolation in Geologic		Formations* Impact Statement on Commercial Waste	000457
Aspects of Disposal of High-Level Waste in Geologic		Formations* Assessment of the Radiological Protection	000264
Occurring Hazardous Materials in Deep Geologic		Formations: A Perspective on the Relative Hazard of	000084
Preliminary *Structural Analysis of Salt Cavities		Formed by Solution Mining: 1. Method of Analysis and	000540
*Importance of the Chemical Species		Formed in the Interaction*	000032
of Solidification Agents used and Products		Formed with Radioactive Wastes at LWR Nuclear Power	000621
Concretes - *Radioactive Waste Fixation in PUETAP (		Formed Under Elevated Temperature and Pressure)	000060
*PIRN Wastes: Permanent Isolation in Rock-		Forming Minerals*	000189
*Hydrothermal Reactivity of Simulated Nuclear Waste		Forms and Water-Catalyzed Waste-Rock Interactions*	000054
*Solid		Forms for Savannah River Plant Radioactive Wastes*	000683
Studies: Battelle Pacific Northwest Laboratory Waste		Forms in Salt* 16. Repository Preconceptual Design	000550
*Radiolytic Gas Production from Tritiated Waste		Forms*	000100
*Natural Glasses: Analogues for Radioactive Waste		Forms*	000111
*Raman Spectra of Cerium Aluminosilicate Waste		Forms*	000113
*Long-Term Elevated Temperature Leaching of Solid Waste		Forms*	000124
*Leach Rate Characterization of Solid Radioactive Waste		Forms*	000286
*Development of Multibarrier Nuclear Waste		Forms*	000665
for Determining Leachabilities of Radioactive Waste		Forms* *Procedure	000084
of Glass and Glass Ceramic Nuclear Waste		Forms* *Characterization	000635

PERMUTED TITLE INDEX

Water-Bearing Horizons for Disposal of Liquid @
Cake to Uranium Hexafluoride. Part 1. The Fluorination-
Shear Waves* @
@An Approach to the
@In-Situ Experiments on Nuclide Migration in
@Flow in
Determination of the Hydraulic Properties of
@Waste Disposal by Shale
@Waste Disposal by Shale
Johnson @Flowability of Bulk Solids Used in Shale
Transport in Vertical Boreholes and Brine Invasion into
Transport in Vertical Boreholes and Brine Invasion into
Transport in Vertical Boreholes and Brine Invasion into
Aquifer. 3. Numerical Simulation of Brine Invasion of
of the Mesozoic Aquifer System Below the Zone of
Reprocessing Waste. 8. @Handling of Spent Nuclear
Consequences from Sabotage of Shipping Casks for Spent
@The Canadian Program for Storage and Disposal of Spent
of the Consequences of Light Water Reactor Spent
@Shipments of Nuclear
@Transportation of Spent
Guides - Fabrication of Light-Water Reactor
Power Industry* @Projections of Spent
@Interim Storage of Spent
General @Engineered Storage for the Disposal of Non-
Packaging, Transportation, and Disposal of Non-
@Problems in the Disposal of Irradiated
- Reprocessing of High-Temperature Gas-Cooled Reactor
Environmental Impact of Waste Effluents in the Nuclear
Environmental Impact of Waste Effluents in the Nuclear
Environmental Impact of Waste Effluents in the Nuclear
Management. @An Analysis of the Back End of the Nuclear
United States and in Doubt from an Integrated Nuclear
@Chemical Engineering Division
@Fusion
and Burial Methods for Radioactive Wastes of
April 1 to @Waste Management Analysis for the Nuclear
for Solid and Liquid Transuranic Waste from the Nuclear
for Solid and Liquid Transuranic Waste from the Nuclear
System for Solid and Liquid TRU Waste from the Nuclear
Disposal in Bedded Salt of LWR Wastes from Alternative
Wastes from the Uranium-Plutonium and Thorium-Uranium
Cell Facilities* @Spent
Decontamination and Melt Densification of Chop-Leach
Monitoring of Technetium 99 and Selenium 79 in Nuclear
of Carbon 14, Krypton 85 and Iodine 129 in Nuclear
Waste Discharges to the Environment from a Nuclear
Highly Active Declad Zircaloy Solid Wastes from a
Disposal Formations in Location of a Large Nuclear-
Term Planning for Management of Aqueous Wastes from
Problems of High-Level and Actinide-Contaminated Spent
of Liquid Metal Fast Breeder Reactor Spent
Level Radioactive Waste Management Program at Nuclear
Valley, @Issues Related to the Closing of the Nuclear
@Fire Protection Device for LMFBR Spent
@Removal of Cesium and Strontium from
Leaking into the Idaho Chemical Processing Plant
@Leak Testing Fuel Stored in the ICPP
Reactor Storage* @Spent
@Tennessee Valley Authority Away-From-Reactor Spent
@Leak Testing
for the Transportation of Radioactive Waste and Spent
of Canisters for Final Disposal of Spent Nuclear
Chemical Processing Plant Waste Canisters and Spent
Concretes - Experimental @Radioactive Waste Fixation in
Strips: Preliminary Results* @
@Leaching of
Facility* @
238 and Curium 242 from a Contaminated Soil as a
@The
@
@Management of Nontritium Radioactive Wastes from
@Evaluation of the Activity Levels in
@Annual Report on DOWT PROJECT AB0115A,
of Cores from Holes P-3 HH-1 through P-5 HH-5, Area
@Thermal Properties of
Transport through Porous Media: A Finite-Element
Nuclear Waste* @Neutron and
Plastic Analysis of Waste-Container Sleeve: 4. Air
Contaminated Hydrogenous Waste* @
@ @Radiolytic
Wastes* @Radiolytic
Plant Waste* @Radiolytic
@Radiolytic
@A Continuous Tritium Monitor for Nuclear Facility Off-
the Subsurface in Russell, @Salt Dissolution in Oil and
@Salt Dissolution in Oil and
Nuclear Fuel Reprocessing and Waste Solidification Off-
Foundational Prerequisites and Practice for Using Deep 000203
Fractionation Process*Fuel Cycle - Conversion of Yellow 000521
Fracture Detection in Crystalline Rock Using Ultrasonic 000481
Fracture Hydrology at Stripa: Preliminary Results* 000398
Fractured Crystalline Rocks* 000276
Fractured Porous Media* 000282
Fractures* of a Potential Size-Effect in Experimental 000090
Fracturing at Oak Ridge National Laboratory* 000213
Fracturing at Oak Ridge National Laboratory* 000214
Fracturing Grouts as Determined by the Jenike and 000681
Fresh Water Aquifer. 1. Numerical Study of Convective- 000272
Fresh Water Aquifer. 2. Experimental Study of 000274
Fresh Water Aquifer. 3. Numerical Simulation of Brine 000275
Fresh Water Aquifers* Brine Invasion into Fresh Water 000275
Fresh Water, Atlantic Coastal Plain, North Carolina 000366
Fuel and Final Storage of Vitrified High-Level 000123
Fuel and High-Level Waste*Calculations of Radiological 000506
Fuel and High-Level Wastes* 000329
Fuel and Plutonium Shipping Package Loss at Sea* 000114
Fuel and Waste...Are They Really Safe?* 000152
Fuel by Barge and By Rail: A Comparison of Risk* 000565
Fuel from Enriched Uranium Dioxide* Low As Practicable* 000652
Fuel to be Discharged by the United States Nuclear 000690
Fuel Assemblies* 000125
Fuel Bearing Components from Nuclear Facilities and 000084
Fuel Bearing Hardware Resulting from Light Water 000651
Fuel Cladding* 000416
Fuel Containing Uranium 233 and Thorium* Fuel Cycle 000501
Fuel Cycle - Conversion of Yellow Cake to Uranium 000521
Fuel Cycle - Reprocessing of High-Temperature Gas- 000501
Fuel Cycle for Use in Establishing "As Low As 000652
Fuel Cycle with Emphasis on High-Level Waste 000321
Fuel Cycle Facility* in Door from the Southeastern 000271
Fuel Cycle Programs Progress Report, January-March 1978* 000336
Fuel Cycle Solid Radioactive Wastes* 000697
Fuel Cycle* @Development of Solidification 000302
Fuel Cycle-Parts 1 and 2: Progress Report for Period 000687
Fuel Cycle: January-March 1978*@Volume Reduction System 000570
Fuel Cycle: July-September 1978*@Volume Reduction System 000571
Fuel Cycle: October-December 1977* @Volume Reduction 000638
Fuel Cycles* @Estimates of Relative Areas for the 000050
Fuel Cycles* Contaminated Spent Fuel Reprocessing 000035
Fuel Handling and Packaging Program. A Survey of Hot 000125
Fuel Hulls* @Chemical 000598
Fuel Reprocessing and Waste Solidification Facility Off- 000598
Fuel Reprocessing and Waste Solidification Off-Gas* 000285
Fuel Reprocessing Plant* @Radioactive 000633
Fuel Reprocessing Plant* @Management of 000097
Fuel Reprocessing Plant* @Importance of Deep Permeable 000291
Fuel Reprocessing Plants* @Long- 000523
Fuel Reprocessing Wastes from the Uranium-Plutonium and 000035
Fuel Rods* of Organic Coolants for the Transportation 000036
Fuel Services* @The High- 000588
Fuel Services, Incorporated, Reprocessing Plant at West 000168
Fuel Shipping Casks* 000568
Fuel Storage Basin Water* 000687
Fuel Storage Basin Water* Source of Radionuclides 000188
Fuel Storage Basin* 000631
Fuel Storage Requirements - The Need for Away-From- 000705
Fuel Storage Studies* 000320
Fuel Stored in the ICPP Fuel Storage Basin* 000631
Fuel* @Logistics Models 000566
Fuel* @Corrosion Resistance 000132
Fuels Preliminary Description and Operating Procedures 000128
FUETAP (Formed Under Elevated Temperature and Pressure) 000060
Full-Scale and Time-Scale Testing Experiments at 000381
Fully Radioactive High-Level Waste Glass* 000358
Function and Licensing Status of the Asse Test Storage 000497
Function of pH and Certain Soil Components: HNO3-NaOH 000289
Fundamentals of Ion Exchange Equilibria* 000676
Fusion Fuel Cycle Solid Radioactive Wastes* 000697
Fusion Power Plants* 000701
Fusion Reactor Blankets, Waste Problem* 000699
FY-1975* 000252
G, Technical Area 54* @Geologic Description 000196
Gable Mountain Basalt Cores Hanford Nuclear Reservation* 000053
Galerkin Model* @Material 000283
Gamma-Ray Sources in Light Water Reactor High-Level 000590
Gap Influence on Hole Closure* @Thermoelastic/ 000180
Gas Generation from Radiolytic Attack of Transuranic- 000091
Gas Generation in Plutonium Contaminated Waste Materials 000083
Gas Production during Long-Term Storage of Nuclear 000577
Gas Production from Concrete Containing Savannah River 000099
Gas Production from Tritiated Waste Forms* 000100
Gas Streams* 000172
Gas Test Holes in Central Kansas, Part 1: Salt Beds in 000879
Gas Test Holes in Kansas* 000878
Gas* of Carbon 14, Krypton 85 and Iodine 129 in 000285

PLYMOUTH TITLE INDEX

Fuel Reprocessing and Waste Solidification Facility Off-	Gas* of Technetium 99 and Selenium 79 in Nuclear	000598
Nuclear Fuel Cycle - Reprocessing of High-Temperature	Gas-Cooled Reactor Fuel Containing Uranium 233 and	000581
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste in Geologic Formations. Volume	000552
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000021
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000022
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000023
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000024
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000237
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000384
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000385
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000545
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000546
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000547
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000548
Support for Generic Environmental Impact Statement (	GEIS): Radioactive Waste Isolation in Geologic	000549
Disposal System* @Operation of the Monticello Nuclear	Generating Plant Liquid and Solid Radioactive Waste	000595
Contaminated Hydrogenous Waste* @Gas	Generation from Radiolytic Attack of Transuranic-	000091
@Radiolytic Gas	Generation in Plutonium Contaminated Waste Materials*	000043
Isolation Safety Assessment Program - Collection and	Generation of Transport Data* No. 45901RE. Waste	000219
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000021
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000022
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000023
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000024
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000237
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000384
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000385
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000545
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000546
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000547
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000548
Radioactive Waste Isolation in @Technical Support for	Generic Environmental Impact Statement (GEIS):	000549
Radioactive Waste in Geologic @Technical Support for	Generic Environmental Impact Statement (GEIS):	000552
Waste Management: Radioactive @Contribution to Draft	Generic Environmental Impact Statement on Commercial	000057
Waste Isolation in Geologic @Technical Support for	Generic Environmental Impact Statement: Radioactive	000075
Waste Isolation in Geologic @Technical Support for	Generic Environmental Impact Statement: Radioactive	000296
Waste Isolation in Geologic @Technical Support for	Generic Environmental Impact Statement: Radioactive	000550
@Radionuclide Migration from Low-Level Waste: A	Generic Overview*	000302
@Selection and Description of	Generic Repository Sites*	000450
Salt Dome Basin* @	Genesis and Growth of Salt Domes in the Northeast Texas	000032
@	Geochemical Behavior of Long-Lived Radioactive Wastes*	000254
Critical Masses from Stored Nuclear Waste in Natural @	Geochemical Constraints on Accumulation of Actinide	000015
Crystalline Rock Repository Media* @Long Term	Geochemical Interactions of High Active Waste with	000157
Disposal in Argillaceous and Crystalline Rocks: Some	Geochemical, Geomechanical, and Hydrogeological	000218
the Stripa Granite: Results and Preliminary @	Geochemistry and Isotope Hydrology of Groundwaters in	000397
@	Geochemistry and Radionuclide Migration*	000269
* @	Geochemistry of Ground Water at the Savannah River Plant	000100
@Hanford Studies in	Geochemistry*	000286
of Plutonium-Release Consequences on the Environmental	Geochemistry* @Analysis	000270
Pilot Plant (WIPP) Site, Southeastern New Mexico.	Geochemistry* Characterization Report, Waste Isolation	000074
Mobility and Retention; Element Chemistry and	Geochemistry: Conclusions and Evaluation* Radionuclide	000217
@	Geohydrologic Studies of Gulf Coast Interior Salt Domes*	000460
@	Geohydrology*	000461
Placement of High-Level Radioactive @Rock Types, Also	Geologic and Hydraulic Settings, Favorable to Deep	000068
Repositories for Radioactive Wastes* @	Geologic and Hydrologic Factors for Locating	000427
Storage Potential of Mesozoic @Evaluation of the	Geologic and Hydrologic Factors Related to the Waste	000365
Realistic Modeling that Allows @Comparison of Some	Geologic and Ocean Disposal Concepts Regarding	000517
@Deep	Geologic Alternative Actions*	000557
@Subsurface Disposal in	Geologic Basins - A Study of Reservoir Strata*	000400
@Economic and Industrial Potential of	Geologic Basins and Reservoir Strata*	000399
through P-3 RM-5, Area G, Technical Area 54* @	Geologic Description of Cores from Holes P-3 RM-1	000196
@The United States Program for the Safety Assessment of	Geologic Disposal of Commercial Radioactive Wastes*	000319
Earth-Science Perspectives* @	Geologic Disposal of High-Level Radioactive Wastes -	000362
@Earth Science Technical Plan for Mined	Geologic Disposal of Radioactive Waste**	000503
of Nuclear Waste Monazites, Ideal Actinide Hosts for	Geologic Disposal* @Synthesis	000440
Selected Segments of the Mesozoic Aquifer System Below @	Geologic Evaluation of Waste-Storage Potential in	000366
from the Appalachian Mountain @Supplementary Report:	Geologic Feasibility of Talc and Serpentinite Bodies	000484
@A Waste Repository in a	Geologic Formation*	000529
of Radiation on the System Liquid Radioactive Wastes -	Geologic Formation* @Influence	000070
@Waste Isolation in	Geologic Formations in the United States of America*	000520
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Baseline Rock Properties--Granite*	000021
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Facility Construction Feasibility	000546
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Stratigraphies of Salt, Granite,	000385
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 1. Executive Summary*	000552
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 10. Repository	000547
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 12. Repository	000548
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 18. Repository	000549
Impact Statement: Radioactive Waste Isolation in	Geologic Formations. Volume 16. Repository	000550
Impact Statement: Radioactive Waste Isolation in	Geologic Formations. Volume 19 - Thermal Analyses*	000075
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 20. Thermomechanical Stress	000024
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 21. Ground Water Movement	000237
Impact Statement: Radioactive Waste Isolation in	Geologic Formations. Volume 23 - Environmental Effluent	000296
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 4. Baseline Rock	000384
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 6. Baseline Rock Properties	000022
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 7. Baseline Rock Properties	000023
Impact Statement (GEIS): Radioactive Waste Isolation in	Geologic Formations. Volume 8. Repository Preconceptual	000545
@Light-Water Reactor Wastes to be Isolated in	Geologic Formations*	000166
@Diffusion of Radionuclides from	Geologic Formations*	000238
@Terminal Storage of Radioactive Wastes in	Geologic Formations*	000425

PERMUTED TITLE INDEX

Term Risk Assessment of Radioactive Waste Disposal in Geologic Formations* ©Long- 000256
of High-Level Radioactive Waste Disposal in Deep Geologic Formations* ©Environmental Aspects 000538
Protection Aspects of Disposal of High-Level Waste in Geologic Formations* Assessment of the Radiological 000264
Waste Management: Radioactive Waste Isolation in Geologic Formations* Impact Statement on Commercial 000057
of Naturally Occurring Hazardous Materials in Deep Geologic Formations: A Perspective on the Relative 000004
Deposits, Clark County, Southeastern Nevada to © Geologic Investigation of the Virgin River Valley Salt 000439
Analysis Methods for Use in Assessing the Safety of the Geologic Isolation of Nuclear Waste* ©Program Scenario 000542
©The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive Waste* 000536
©Ohio - Geologic Isolation Implications* 000220
©Long-Term Safety Assessment of Geologic Isolation Repositories* 000369
Level Waste Canister Corrosion Studies Pertinent to Geologic Isolation* ©High- 000103
©Microstructural Interactions of Geologic Media with Waste Radionuclides* 000038
©The Rock-Salt Approach to Nuclear Waste Disposal in Geologic Media* 000017
©Sorption and Migration of Radionuclides in Geologic Media* 000204
©Transport Properties of Nuclear Waste in Geologic Media* 000298
Eh During Adsorption of Multivalent Elements by Geologic Media*©Electron Transfer Method of Controlling 000070
1977* ©Studies of Nuclear-Waste Migration in Geologic Media: Annual Report, November 1976-October 000299
Neptunium 237* © Geologic Migration Potentials of Technetium 99 and 000011
Terminal Storage Program* ©Preliminary Geologic Site Selection Factors For the National Waste 000080
of Wastes and Containers During Retrievable Geologic Storage* ©Maximum Acceptable Temperatures 000121
Texas Salt Dome Basin to Investigate Their Suitability © Geologic Study of the Interior Salt Domes of Northeast 000438
References to the Subsurface Disposal of Radioactive © Geologic Summary of the Appalachian Basin with 000379
©Nuclear Waste Disposal: The Geological Aspects* 000375
Pilot Plant (WIPP) Site, Southeastern New Mexico. © Geological Characterization Report, Waste Isolation 000073
Pilot Plant (WIPP) Site, Southeastern New Mexico. © Geological Characterization Report, Waste Isolation 000074
Pilot Plant (WIPP) Site, Southeastern New Mexico* © Geological Characterization Report, Waste Isolation 000338
Pilot Plant (WIPP) Site, Southeastern New Mexico. © Geological Characterization Report, Waste Isolation 000053
Pilot Plant (WIPP) Site, Southeastern New Mexico. Site © Geological Characterization Report, Waste Isolation 000054
Pilot Plant (WIPP) Site, Southeastern New Mexico. © Geological Characterization Report, Waste Isolation 000055
Pilot Plant (WIPP) Site, Southeastern New Mexico. © Geological Characterization Report, Waste Isolation 000056
© Geological Criteria for Radioactive Waste Repositories* 000368
Radioactive Wastes* © Geological Criteria for Repositories for High-Level 000025
Economic Cooperation and Development Countries Forum Geological Disposal* ©Organization for 000404
©Radioactive Waste Disposal in Geological Formations* 000402
of Solid High-Level and Alpha-Bearing Wastes in Geological Formations* ©Site Selection for Repositories 000013
Waste Disposal* ©Pollucite and Its Alteration on Geological Occurrences and in Deep-Burial Radioactive 000105
Waste Repository in a Bedded Salt ©An Analysis of the Geological Stability of a Hypothetical Radioactive 000562
Reactor Products and Applications to the Problem of Geological Storage of Nuclear Wastes* Paths for Ohio 000236
Relationships in Salt Repositories* ©United States Geological Survey Studies of Physical-Chemical 000079
©Ion Exchange Research by the United States Geological Survey* 000064
©Adsorption Studies by the U.S. Geological Survey* 000576
© Geology and Description of the Lyons Mine* 000320
Reservation of the U.S. Atomic Energy Commission, © Geology and Ground-Water Characteristics of the Hanford 000189
©Environmental Geology in Development of Nuclear Energy* 000452
Valley Anticline, Grand County, Utah* ©Subsurface Geology of a Potential Waste Emplacement Site, Salt 000411
©Regional Geology of the Northeast Texas Salt Dome Basin* 000412
© Geology of the 241-AX Tank Farm* 000193
© Geology of the 241-B Tank Farm* 000194
© Geology of the 241-BY Tank Farm* 000195
Dome Basin* © Geology of Certain Domes of the Northeast Texas Salt- 000353
©Sedimentary History and Economic Geology of San Juan Basin, New Mexico and Colorado* 000443
© Geology of Subsurface Waste Disposal in Michigan Basin* 000368
©Regional Geology* 000405
Pilot Plant (WIPP) Site, Southeastern New Mexico. Site Geology* Characterization Report, Waste Isolation 000054
Plant (WIPP) Site, Southeastern New Mexico. Regional Geology* Characterization Report, Waste Isolation Pilot 000053
Argillaceous and Crystalline Rocks: Some Geochemical, Geomechanical, and Hydrogeological Questions* in 000218
© Geomechanics* 000072
Plant (WIPP) Site in ©Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot 000241
Underlying the Migration of Radionuclides in Geomedia: Ohio and Other Examples* Considerations 000231
©Interaction of Waste Radionuclides with Geomedia: Program Approach and Progress* 000071
their Exploitation for Ultimate ©The Waterproof Geometry of Salt Domes, An Important Safety Aspect for 000407
Strips* ©Preliminary Report on Geophysical and Mechanical Borehole Measurements at 000437
Part of the Atlantic Coastal Plain, South Carolina and Georgia* Potential of Nonsaline Aquifers in the Southern 000365
at Western New York Nuclear Service Center, West © Geotechnical Analysis of Soil Samples from Test Trench 000173
History of Shales* © Geothermometry Methods for Determining the Thermal 000087
Wastes Produced in Nuclear Installations in the German Democratic Republic* ©Disposal of Radioactive 000450
Study, Radioactive Wastes in the Federal Republic of Germany. Vol 8. The Law Concerning the Disposal of 000495
Program on Waste Disposal in the Federal Republic of Germany* ©The Research and Development 000461
of Radioactive Wastes in the Federal Republic of Germany* the Federal Republic of Germany: Accumulation 000496
Federal ©Radioactive Wastes in the Federal Republic of Germany: Accumulation of Radioactive Wastes in the 000496
©Radioactive Waste Disposal in Antarctica: Some Glaciological Aspects* 000441
©Characterization of Glass and Glass Ceramic Nuclear Waste Forms* 000631
Highly Active Radioactive Wastes* © Glass as the First Barrier for Long-Term Storage of 000128
©The Leaching of Radioactivity from Highly Radioactive Glass Blocks Buried Below the Water Table: Fifteen 000685
©Characterization of Glass and Glass Ceramic Nuclear Waste Forms* 000635
Plant Waste* ©Development of Glass Compositions for Immobilization of Savannah River 000658
©Development of Glass Compositions for Immobilization of SRP Waste* 000656
©Effect of Internal Alpha Radiation on Borosilicate Glass Containing Savannah River Plant Waste* 000099
©Volatility of a Zinc Borosilicate Glass Containing Simulated High-Level Radioactive Waste* 000608
©Hydrothermal Glass Reactions in Salt Brine* 000155
©Chemistry of Nuclear Waste Glass Reactions: Problems and Potential of Prediction* 000093
and Borosilicate ©Fixation of Radioactive Waste in Glass. Part 2. The Experimental Evaluation of Phosphate 000108
©Bonding of Metal Oxides in Sodium Silicate Glass* 000126
©Leaching of Fully Radioactive High-Level Waste Glass* 000358
©High-Level Waste Glass* 000643
Behavior in a Zinc Borosilicate Nuclear Waste Glass* ©Devitrification 000679
of Actinides from Simulated Reactor Waste Glass* ©Leaching Characteristics 000153
High Level Radioactive Waste (HLW) Borosilicate Glass? ©Preparation and Characterization of an Improved 000605

PERMUTED TITLE INDEX

*Interaction of a
*Preliminary Corrosion Test of a
of Leached Surface Layers on Simulated High-Level Waste
*Development and Radiation Stability of
of Radionuclide Leachabilities with Microstructures of
*Viscosity of
Report on the Characterization of High-Level Waste
Alpha-Decay and its Significance for Radioactive Waste
Experimental Evaluation of Phosphate and Borosilicate
*Natural
Waste Problems* *Let's
* 000159
Brine Inclusions *Effects of Temperature, Temperature
Long-Term Isolation of Solid Radioactive Waste in Fine-
*Fine-
Waste Emplacement Site, Salt Valley Anticline,
*Analysis of Some Tracer Runs in
Cesium, Barium, Cerium, *Sorption-Desorption Studies on
*The Mechanical Properties of Stripa
*A Pilot Heater Test in the Stripa
Volume 10. Repository Preconcept' Design Studies:
in Geologic Formations. Baseline Rock Properties--
in Geologic Formations. Stratigraphics of Salt,
and Isotope Hydrology of Groundwaters in the Stripa
Studies at Elevated Temperature and Pressure of
*Radioactive Waste "
Marine Sediment on the Accumulation of Radionuclides by
*A Review of Radioactive Waste Disposal to the
Natural Nuclear Reactor Concerning the Evolution in the
*Above-
Pertinent to the Migration of Radionuclides in
*Geochemistry of
*Kinetic Model for Silicate Rock-
Waste Isolation in Geologic Formations. Volume 21.
of the U.S. Atomic Energy Commission, *Geology and
*Burial
*Nuclide Transport by
*Hanford
Radioactive Waste Incidents* *Hanford
Preliminary *Geochemistry and Isotope Hydrology of
90 and Cobalt 60 Leached from Matrix of Hardened Cement
*Flowability of Bulk Solids Used in Shale Fracturing
Calibration from Detroit Mine Experiment* *
Basin* *Genesis and
*
Stress Analysis and Development of Thermal Loading
Cycle for Use in Establishing "As Low As Practicable"
*Geohydrologic Studies of
Radioactive Wastes* *An Investigation of the Utility of
*Compilation of
*Technical Challenges in the Immobilization of
*
Radioactive Waste Incidents* *
*
*Thermal Properties of Gable Mountain Basalt Cores
and Process Development, July 1977 through *Rockwell
*Environmental Assessment Related to
*A Review of Studies on Soil-Waste Relationships on the
*Geology and Ground-Water Characteristics of the
Term Management of Defense High-Level Radioactive Waste
*Fixation of Radioisotopes on
*
Affecting Sorption and Transport of Radionuclides in
*Operating Practices and Experiences at
*High-Level Waste Leakage from the 241-T-106 Tank at
*Mineral Reaction Work at
A Review of Radioactive Waste Disposal to the Ground at
Design for Retrieval of Solidified High-Level Wastes at
(B and D) of Retired Contaminated Facilities at
Underlying Liquid Waste Disposal Facilities at
*Solidification and Storage of
Radioactive Wastes* *An Appraisal of
Radioactive Wastes* *An Appraisal of
Considerations for Radioactive Waste Repositories in
*In Situ Heating Experiments in
137, Strontium 90 and Cobalt 60 Leached from Matrix of
Transportation, and Disposal of Non-Fuel Bearing
of an Improved High Level Radioactive Waste (
Impact Program (AIIP), Part I. Carcinogenic
Deep Geologic Formations: A Perspective on the Relative
*Effect of Actinide Removal on the Long-Term
Perspective on the *Survey of Naturally Occurring
*
*Public
Wastes in Clay-Plus Sintered*
Waste Storage Tanks* *
*Improved
Glass-Based Nuclear Waste Simulant and Natural Rock* 000040
Glass-Ceramic Candidate for a Nuclear Waste Canister* 000151
Glasses by Sputter-Induced Optical Emission* 000119
Glasses for Highly Radioactive Wastes* 000508
Glasses Containing Savannah River Plant Waste* 000620
Glasses Containing Simulated Savannah River Plant Waste* 000657
Glasses* *Annual 000669
Glasses* *Helium Formation from 000131
Glasses* of Radioactive Waste in Glass. Part 2. The 000108
Glasses: Analogues for Radioactive Waste Forms* 000111
Go Slowly on a Nuclear Power Program Until We've Solved 000527
Gold and Selective Storage Solves Nuclear Waste Problem* 000159
Gradients, Stress, and Irradiation on Migration of 000041
Grained Deposits* *A Concept for 000376
Grained Sheet Silicate Rocks* 000482
Grand County, Utah* *Subsurface Geology of a Potential 000411
Granite Rock Using a Fissure Model* 000287
Granite. 1. Initial Studies of Strontium, Technetium, 000028
Granite* 000469
Granite* 000535
Granite* Waste Isolation in Geologic Formations. 000587
Granite* Statement (GEIS): Radioactive Waste Isolation 000021
Granite, Shale, and Basalt* Radioactive Waste Isolation 000365
Granite: Results and Preliminary Interpretation* 000397
Granitic Rocks* *Experimental Permeability 000069
Graveyards"* 000085
Green Alga (Ulva pertusa)* *Influences of 000285
Ground at Hanford* 000191
Ground of Radioactive Waste over a Long Period*the Chlo 000247
Ground Storage Facilities for Vitrified Highly Active 000573
Ground Water at the Nevada Test Site - Part 2: 000163
Ground Water at the Savannah River Plant* 000180
Ground Water Interaction* 000088
Ground Water Movement and Nuclide Transport*Radioactive 000237
Ground-Water Characteristics of the Hanford Reservation 000189
Grounds* 000168
Groundwater in Swedish Bedrock* 000001
Groundwater Scenario Studies* 000222
Groundwater Transport Estimates of Hypothetical 000221
Groundwaters in the Stripa Granite: Results and 000397
Grout* of Specimen on Amounts of Cesium 137, Strontium 000639
Grouts as Determined by the Jenike and Johanson Method* 000681
Growth and Shape of Salt Cavity--Computer Model and 000462
Growth of Salt Domes in the Northeast Texas Salt Dome 000832
Guide to Radioactive Waste Management Literature* 000323
Guidelines* Formations. Volume 20. Thermomechanical 000028
Guides - Fabrication of Light-Water Reactor Fuel from 000652
Gulf Coast Interior Salt Domes* 000460
Gulf Coast Salt Domes for the Storage or Disposal of 000828
Hanford Corrosion Studies* 000129
Hanford Defense Waste* 000627
Hanford Groundwater Scenario Studies* 000222
Hanford Groundwater Transport Estimates of Hypothetical 000221
Hanford Mineral Exchange Program* 000689
Hanford Nuclear Reservation* 000053
Hanford Operations Quarterly Report, Process Technology 000335
Hanford Radioactive Waste Burial* 000268
Hanford Reservation from 1944 to 1967* 000197
Hanford Reservation of the U.S. Atomic Energy 000189
Hanford Reservations, Richland, Washington* for Long- 000387
Hanford Soil Types* 000183
Hanford Studies in Geochemistry* 000286
Hanford Subsoils* *Variables 000358
Hanford* 000170
Hanford* 000190
Hanford* 000572
Hanford* * 000191
Hanford* *System 000212
Hanford* *Disposition 000175
Hanford*Characterization of Actinide-Bearing Sediments 000290
Hanford's High-Level Radioactive Liquid Wastes* 000669
Hard Rock for Potential Underground Repositories of 000018
Hard Rock for Potential Underground Repositories of 000380
Hard Rock. Part 1 and 2**Mechanical and Thermal Design 000019
Hard Rock: Their Objectives and Design* 000382
Hardened Cement Grout* of Specimen on Amounts of Cesium 000639
Hardware Resulting from Light Water Reactor Operations* 000651
HAW) Borosilicate Glass* and Characterization 000605
Hazard from Low-Level, Low-Rate Radiation* Radiological 000253
Hazard of Deep Burial of Nuclear Wastes* Materials in 000088
Hazard of High-Level Waste* 000585
Hazardous Materials in Deep Geologic Formations: A 000088
Hazards to Workers* 000383
Hazards Analysis* 000530
Health Implications of Radioactive Waste Releases* 000339
Heat Experiments for Self Fixation of Radioactive 000617
Heat Pipe Cooling System for Underground, Radioactive 000107
Heat Transfer from Radioactive Waste Canisters* 000119

PERMUTED TITLE INDEX

*Non-Radioactive Operation Experience with a Joule	Heated Ceramic Melter for Vitrification of High-Level 000686
*A Review of the Heat Surface	Heater Experiment at Oak Ridge, Tennessee* 000821
*Data Acquisition, Handling, and Display for the	Heater Experiments at Stripa* 000056
*	Heater Experiments in the Climax Stock, Nevada Test Site 000846
*Theoretical Temperature Fields for the Stripa	Heater Project, Volume I* 000378
*A Pilot	Heater Test in the Stripa Granite* 000535
*Electrical	Heaters for Thermo-Mechanical Tests at the Stripa Mine* 000538
*Full-Scale and Time-Scale	Heating Experiments at Stripa: Preliminary Results* 000381
Design* *In Situ	Heating Experiments in Hard Rock: Their Objectives and 000382
for Radioactive Waste Glasses* *	Helium Formation from Alpha-Decay and its Significance 000131
*Nuclear Waste: Twenty-Five Years	Hence ... And Beyond* 000372
Fuel Cycle - Conversion of Yellow Cake to Uranium	Hexafluoride. Part 1. The Fluorination-Fractionation 000521
Mexico and Colorado* *Sedimentary	History and Economic Geology of San Juan Basin, New 000883
While in Air and Water Storage* *	History of Prototype High Level Waste Canister SS-9 000101
*Geothermometry Methods for Determining the Thermal	History of Shales* 000007
Soil as a Function of pH and Certain Soil Components:	UO_2-$BaCO_3$ System*238 and Curium 242 from a Contaminated 000288
Repository* *The Utilization of the	HDL Data Structure in Consequence Analysis for WIPP 000223
of Waste-Container Sleeve: 4. Air Gap Influence on	Hole Closure* *Thermoelastic/Plastic Analysis 000180
*Deep Exploratory Test	Holes for Waste Repository Investigations* 000826
Subsurface in *Salt Dissolution in Oil and Gas Test	Holes in Central Kansas, Part 1: Salt Beds in the 000879
*Salt Dissolution in Oil and Gas Test	Holes in Kansas* 000878
Sa* *Geologic Description of Cores from	Holes P 3 MB-1 through P-3 MB-5, Area G, Technical Area 000196
*Mathematical Modelling of Radionuclide Migration in a	Homogeneous Clayey Formation (Application to the Case 000291
Prerequisites and Practice for Using Deep Water-Bearing	Horizons for Disposal of Liquid Radioactive Wastes* 000203
*Studies of Concrete as a	Host for Savannah River Plant Radioactive Waste* 000672
of Alabama, Mississippi and Tennessee as Potential	Host Rocks for Underground Emplacement of Waste* Parts 000830
*Synthesis of Nuclear Waste Monazites, Ideal Actinide	Hosts for Geologic Disposal* 000680
*Spent Fuel Handling and Packaging Program. A Survey of	Hot Cell Facilities* 000125
Containment of Radioactive Waste* *	Hot Isostatic Pressing for the Consolidation and 000620
*Corrosion of Metals in a	Hot, High Pressure Brine Environment* 000130
and Melt Densification of Chop-Leach Fuel	Hulls* *Chemical Decontamination 000588
Station* *Technological Facilities at the New	Hungarian Radioactive Waste Handling and Disposal 000593
Size-Effect in Experimental Determination of the	Hydraulic Properties of Fractures* of a Potential 000090
Level Radioactive Wastes**Rock Types, Also Geologic and	Hydraulic Settings, Favorable to Deep Placement of High- 000068
*Some Properties of	Hydrogen-Based Clinoptilolite* 000678
from Radiolytic Attack of Transuranic-Contaminated	Hydrogenous Waste* *Gas Generation 000091
Crystalline Rocks: Some Geochemical, Geomechanical, and	Hydrogeological Questions* Disposal in Argillaceous and 000219
Nuclear Waste Repository, Eddy *Review and Analysis of	Hydrologic Conditions Near the Site of a Potential 000831
Radioactive Wastes* *Geologic and	Hydrologic Factors for Locating Repositories for 000827
Potential of Mesozoic *Evaluation of the Geologic and	Hydrologic Factors Related to the Waste Storage* 000365
Cavity and Chimney Produced by the Camrikin Nuclear *	Hydrologic Processes and Radionuclide Distribution in a 000377
*An Approach to the Fracture	Hydrology at Stripa: Preliminary Results* 000756
Results and Preliminary *Geochemistry and Isotope	Hydrology of Groundwaters in the Stripa Granite: 000357
*	Hydrology of Some Deep Mines in Precambrian Rocks* 000889
Station, Idaho, An Interim Report* *	Hydrology of Waste Disposal National Reactor Testing 000818
Pilot Plant (WIPP) Site, Southeastern New Mexico.	Hydrology* Characterization Report, Waste Isolation 000656
*Studies on the	Hydrolysis Reactions of Metal Ions* 000064
*	Hydrothermal Glass Reactions in Salt Brine* 000155
*Fixation of Radioactive Waste by	Hydrothermal Reactions with Clays* 000575
Forms and Water-Catalyzed Waste-Rock Interactions* *	Hydrothermal Reactivity of Simulated Nuclear Waste 000158
Radioactive Wastes: Flood Basalt* *	Hypothetical Prototype Sites of Repositories for 000846
*Hanford Groundwater Transport Estimates of	Hypothetical Radioactive Waste Incidents* 000221
Salt *An Analysis of the Geological Stability of a	Hypothetical Radioactive Waste Repository in a Bedded 000562
Study on the International Atomic Energy Agency (	IAEA) Definition of High-Level Radioactive Waste 000189
Repository in Antarctica* *Putting Radioactive Waste on	Ice - A Proposal for an International Radionuclide 000842
*Vitrification of High-Level	ICPP Calcined Wastes* 000602
*Leak Testing Fuel Stored in the	ICPP Fuel Storage Basin* 000631
* *Primary Source of Radionuclides Leaking into the	Idaho Chemical Processing Plant Fuel Storage Basin Water 000188
Waste *Corrosion Evaluation of Stainless Steels in	Idaho Chemical Processing Plant High Level Radioactive 000117
Spent Pools *Neutron Interrogator Assay System for the	Idaho Chemical Processing Plant Waste Canisters and 000178
*Development of a Pelleted Waste Form for High-Level	Idaho Chemical Processing Plant Zirconia Wastes* 000625
*Reprocessing Facilities,	Idaho Chemical Processing Plant* 000623
of Stored High-Level Radioactive Calcined Waste* at	Idaho Chemical Processing Plant* *Sampling 000158
from the Second Calcined Solids Storage Facility at	Idaho Chemical Processing Plant* Calcine Retrieved 000208
Term Management of Defense High-Level Radioactive Waste	Idaho Chemical Processing Plant, Idaho Falls, Idaho* 000186
Radioactive Waste Idaho Chemical Processing Plant,	Idaho Falls, Idaho* Management of Defense High-Level 000186
Site 16, and Leshi Range)* *Present Environment of the	Idaho National Engineering Laboratory (Including RWMC, 000207
Management* *Identification of Alternatives for	Idaho National Engineering Laboratory Transuranic Waste 000175
of Alternatives for Long-Term Management of Stored	Idaho National Engineering Laboratory Transuranic Waste* 000180
*Retrieval of Buried Transuranic Waste at the	Idaho National Engineering Laboratory* 000188
*Decontamination and Decommissioning, Long Range Plan,	Idaho National Engineering Laboratory* 000693
Storage of Intermediate-Level Transuranic Waste at the	Idaho National Engineering Laboratory* and Interim 000187
*Evolving Retrieval, Processing, and Disposal on the	Idaho National Engineering Laboratory* *Alternatives 000182
for the Radioactive Waste Management Complex at the	Idaho National Engineering Laboratory* Review Document 000176
near a Transuranic Storage Area in Southeastern	Idaho* *Plutonium and Americium Contamination 000181
Waste Idaho Chemical Processing Plant, Idaho Falls,	Idaho*Term Management of Defense High-Level Radioactive 000186
of Waste Disposal National Reactor Testing Station,	Idaho, An Interim Report* *Hydrology 000818
*Synthesis of Nuclear Waste Monazites,	Ideal Actinide Hosts for Geologic Disposal* 000680
Engineering Laboratory Transuranic Waste Management* *	Identification of Alternatives for Idaho National 000175
99 and Neptunium 237 in Contact with Unweathered	Igneous Rocks* *Chemistry of Technetium 000229
*	Illite: A Model Based on the Sorption Behavior of Cesium 000013
1977)* *Waste Incineration and	Immobilization for Nuclear Facilities (April-September 000616
*Technical Challenges in the	Immobilization of Hanford Defense Waste* 000627
*U.S. Program for the	Immobilization of High-Level Nuclear Wastes* 000588
*Development of Glass Compositions for	Immobilization of Savannah River Plant Waste* 000658
Level Liquid Wastes by Solidification in Cements* *	Immobilization of Strontium and Cesium in Intermediate- 000668
*Development of Glass Compositions for	Immobilization of SRP Waste* 000656
Waste Treatments Costs and the Environmental	Impact of Waste Effluents in the Nuclear Fuel Cycle - 000501

Radioactive Waste Treatment Costs and the Environmental
Radioactive Waste Treatment Costs and the Environmental
in the Waste Isolation Pilot Plant @Potential Microbial
in the Waste Isolation Pilot Plant @Potential Microbial
in the Waste Isolation Pilot Plant @Potential Microbial
@Potential Microbial
@Correlation of Environmental
Radioactive Waste Management*@Technology Assessment and
Low-Level, Low-Rate Radiation*@The Argonne Radiological
@Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
@Contribution to Draft Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
Geologic @Technical Support for Generic Environmental
@Environmental
@Nuclear Waste: Increasing Scale and Sociopolitical
Mass Transport Model: Theory and Numerical
Method for Radioactive Waste Management. Volume 2.
@Requirements for
@Metamict Mineral Alteration: An
@P-T X Relations of Anhydrite and Brine and Their
@Public Health
@Oklo - Geologic Isolation
Facilities* @Survey of Active and
@
Transport Estimates of Hypothetical Radioactive Waste
Production, and Reprocessing @Status Report, Waste
April-September 1977)* @Waste
Defense Waste, July-December 1977* @Fluidized Bed
@TRU Waste Cyclone Drum
@Operation of Low-Level Radioactive Waste
Monitoring Techniques for a Transuranic Waste
Stress, and Irradiation on Migration of Brine
Problems of Siting the @Application of Studies of Fluid
@Paleozoic Stratigraphy of Two Areas in Southwestern
Strata* @Economic and
@Subsurface Disposal of Liquid
Appalachian Basin in New @Possibilities for Disposal of
to be Discharged by the United States Nuclear Power
Radionuclides by Green Alga (Ulva pertusa)* @
Excavation Technology* @
in Ground Water at the Nevada Test Site - Part 2: @
@Radiation Doses to Rodents
@
Plastic Analysis of Waste-Container Sleeve: 1.
and Pressure) Concretes - Experimental Program and
Barium, @Sorption-Desorption Studies on Granite. 1.
@Dispersion Relations in
Disposal of Liquid Industrial Wastes by Deep-Well
@
@Disposal of Radioactive Wastes Produced in Nuclear
on Treating Radioactive Wastes from Reprocessing
from the Southeastern United States and in Deep from an
on Two United States Subsoils from Differing Weathering
Americium 241 on Two Subsoils from Differing Weathering
Natural Rock* @
with the Waste Isolation Pilot Plant (WIPP) Site in @
Program Approach and Progress* @
Processes Influencing Radionuclide @Radionuclide
Waste Burial Site* @
@Importance of the Chemical Species Formed in the
@Kinetic Model for Silicate Rock-Ground Water
@Waste-Rock
@
@State/Federal Licensing
@Microstructural
Repository Media* @Long Term Geochemical
@
@Waste-Rock
Nuclear Waste Forms and Water-Catalysed Waste-Rock
Numerical Model for Simulating the Deformation @
@Early Waste Retrieval
Disposal National Reactor Testing Station, Idaho, An
at the Idaho National @Transport, Handling, and
@
to Investigate Their Suitability @Geologic Study of the
Impact of Waste Effluents in the Nuclear Fuel Cycle - 000521
Impact of Waste Effluents in the Nuclear Fuel Cycle for 000652
Impact on Transuranic Wastes Under Conditions Expected 000007
Impact on Transuranic Wastes Under Conditions Expected 000008
Impact on Transuranic Wastes Under Conditions Expected 000009
Impact on Transuranic Wastes* 000224
Impact with Cost of Waste Treatment* 000500
Impact Analysis of Separation Methods Applied to 000601
Impact Program (ARIP), Part I. Carcinogenic Hazard from 000253
Impact Statement (GEIS): Radioactive Waste in Geologic 000552
Impact Statement (GEIS): Radioactive Waste Isolation in 000021
Impact Statement (GEIS): Radioactive Waste Isolation in 000022
Impact Statement (GEIS): Radioactive Waste Isolation in 000023
Impact Statement (GEIS): Radioactive Waste Isolation in 000024
Impact Statement (GEIS): Radioactive Waste Isolation in 000237
Impact Statement (GEIS): Radioactive Waste Isolation in 000384
Impact Statement (GEIS): Radioactive Waste Isolation in 000385
Impact Statement (GEIS): Radioactive Waste Isolation in 000545
Impact Statement (GEIS): Radioactive Waste Isolation in 000546
Impact Statement (GEIS): Radioactive Waste Isolation in 000547
Impact Statement (GEIS): Radioactive Waste Isolation in 000548
Impact Statement (GEIS): Radioactive Waste Isolation in 000549
Impact Statement on Commercial Waste Management: 000857
Impact Statement: Radioactive Waste Isolation in 000675
Impact Statement: Radioactive Waste Isolation in 000296
Impact Statement: Radioactive Waste Isolation in 000550
Impact* 000238
Impacts* 000510
Implementation (Discrete-Parcel-Random-Walk Version)* 000215
Implementation for Terminal Storage in Reference 000279
Implementation* 000348
Implication for Radioactive Waste Disposal* 000030
Implications for the Suitability of Anhydrite as a 000016
Implications of Radioactive Waste Releases* 000339
Implications* 000220
Inactive Mines for Possible Use as In Situ Test 000378
Incentives for Partitioning High-Level Waste* 000233
Incidents* @Hanford Groundwater 000221
Incineration and Fixation for Waste Management, 000490
Incineration and Immobilization for Nuclear Facilities (000616
Incineration System for U.S. Department of Energy 000349
Incinerator and Treatment System: July-September, 1978* 000622
Incinerator* 000588
Incinerator* @Selection of a Criticality 000650
Inclusions in a Salt Repository* Temperature Gradients, 000041
Inclusions in Permian Salado Salt, New Mexico, to 000072
Indiana* 000393
Industrial Potential of Geologic Basins and Reservoir 000399
Industrial Wastes by Deep-Well Injection* 000480
Industrial Wastes in Subsurface Rocks on North Flank of 000429
Industry* @Projections of Spent Fuel 000690
Influences of Marine Sediment on the Accumulation of 000285
Information Base for Waste Repository Design, Volume 6. 000543
Information Pertinent to the Migration of Radionuclides 000163
Inhabiting a Radioactive Waste Receiving Area* 000283
Initial Drum Retrieval Final Report* 000185
Initial Estimates of Loading on the Sleeve* 000067
Initial Results* (Formed Under Elevated Temperature 000060
Initial Studies of Strontium, Technetium, Cesium, 000028
Injection Disposal of Radioactive Wastes* 000266
Injection* @Subsurface 000480
Installation of Experimental Equipment* 000539
Installations in the German Democratic Republic* 000450
Installations* @Studies 000625
Integrated Nuclear Fuel Cycle Facility* in Deer 000277
Intensity Areas* 237, and Technetium 99 Sorption 000291
Intensity Areas* of Technetium 99, Neptunium 237, and 000294
Interaction of a Glass-Based Nuclear Waste Simulant and 000040
Interaction of Radionuclides with Geomedia Associated 000241
Interaction of Waste Radionuclides with Geomedia: 000071
Interaction with Soil and Rock Media. Volume 1: 000217
Interaction Between Processing Method and Radioactive 000463
Interaction* 000032
Interaction* 000088
Interactions and Bedrock Reactions* 000089
Interactions between Nuclear Waste and Surrounding Rock* 000282
Interactions for Radioactive Waste Repositories* 000504
Interactions of Geologic Media with Waste Radionuclides* 000038
Interactions of High Active Waste with Crystalline Rock 000157
Interactions of Simulated Waste Radionuclides and Rocks* 000039
Interactions* 000442
Interactions* @Hydrothermal Reactivity of Simulated 000054
Interim Report on Development of a Semi-Empirical 000020
Interim Report* 000186
Interim Report* @Hydrology of Waste 000418
Interim Storage of Intermediate-Level Transuranic Waste 000187
Interim Storage of Spent Fuel Assemblies* 000125
Interior Salt Domes of Northeast Texas Salt Dome Basin 000438

PERMUTED TITLE INDEX

@Geohydrologic Studies of Gulf Coast	
Cements* @In [illegible]ilization of Strontium and Cesium in	
National @Transport, Handling, and Interim Storage of	
Containing Savannah River Plant Waste* @Effect of	
High-Level Radioactive Waste @A Critical Study on the	
Wastes* @Antarctica - A Potential	
@Putting Radioactive Waste on Ice - A Proposal for an	
@	
in the Stripa Granite: Results and Preliminary	
Processing Plant Waste Canisters and Spent @Neutron	
@Polonium and Plutonium in An	
of Salt Transport in Vertical Boreholes and Brine	
of Salt Transport in Vertical Boreholes and Brine	
of Salt Transport in Vertical Boreholes and Brine	
Fresh Water Aquifer. 3. Numerical Simulation of Brine	
Salt Deposits, Clark County, Southeastern Nevada to	
Salt Domes of Northeast Texas Salt Dome Basin to	
@Methodology for the Determination of Environmental	
the Continuous Monitoring of Carbon 14, Krypton 85 and	
@The Disposal of	
Nuclear Waste* @	
@The Fundamentals of	
@	
Survey* @	
@The Design of Systems for Handling Radioactive	
@Contamination of Soils: Behavior of Radioactive	
@Studies on the Hydrolysis Reactions of Metal	
@Problems in the Disposal of	
of Temperature, Temperature Gradients, Stress, and	
Rock Salt Utilizing Measurements Made During Electron	
Produced by the Cannikin Nuclear Explosion, Amchitka	
@Light-Water Reactor Wastes to be	
of America* @Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Environmental Impact Statement: Radioactive Waste	
Environmental Impact Statement: Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Environmental Impact Statement: Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
Impact Statement (GEIS): Radioactive Waste	
on Commercial Waste Management: Radioactive Waste	
@PIRM Wastes: Permanent	
Methods for Use in Assessing the Safety of the Geologic	
of Borehole Plugging and Shaft Sealing for Geologic	
Deposits* @A Concept for Long-Term	
@Corrosion Considerations for Nuclear Waste	
@Description of Waste	
in Salt Domes and How It Affects a Radioactive Waste	
@Effluents from the Waste	
@Oklo - Geologic	
Site, Southeastern New Mexico. Special Studies of Waste	
Radionuclides with Geomedia Associated with the Waste	
Mexico. @Geological Characterization Report, Waste	
Mexico. @Geological Characterization Report, Waste	
Mexico* @Geological Characterization Report, Waste	
Mexico. @Geological Characterization Report, Waste	
Mexico. Site @Geological Characterization Report, Waste	
Mexico. @Geological Characterization Report, Waste	
Mexico. @Geological Characterization Report, Waste	
Wastes Under Conditions Expected in the Waste	
Wastes Under Conditions Expected in the Waste	
Wastes Under Conditions Expected in the Waste	
Waste Storage Under Conditions Expected in the Waste	
@New Mexico Waste	
to Estimate Migration Potential at the Proposed Waste	
@Preliminary Safety Assessment of the Waste	
High-Level Waste* @Waste	
@Forced-Air Cooling of a Waste	
@Environmental Monitoring at the Waste	
@Revised Concept for the Waste	
Salt, New Mexico, to Problems of Siting the Waste	
of RAC-Wastes for the Sandia Laboratories Waste	
@Basalt Waste	
@Basalt Waste	
@Basalt Waste	
@Basalt Waste	
@Basalt Waste	
@Waste-	
@Long-Term Safety Assessment of Geologic	
Control and Accountability Procedures for a Waste	

Interior Salt Domes*	000460
Intermediate-Level Liquid Wastes by Solidification in	000664
Intermediate-Level Transuranic Waste at the Idaho	000187
Internal Alpha Radiation on Borosilicate Glass	000099
International Atomic Energy Agency (IAEA) Definition of	000188
International Burial Area for High-Level Radioactive	000352
International Radionuclide Depository in Antarctica*	000492
International Waste-Management Symposium*	000522
Interpretation* and Isotope Hydrology of Groundwaters	000397
Interrogator Assay System for the Idaho Chemical	000134
Intertidal Food Chain*	000235
Invasion into Fresh Water Aquifer. 1. Numerical Study	000273
Invasion into Fresh Water Aquifer. 2. Experimental	000274
Invasion into Fresh Water Aquifer. 3. Numerical	000275
Invasion of Fresh Water Aquifers* Brine Invasion into	000275
Investigate their Suitability for Possible Storage of	000439
Investigate Their Suitability for Possible Storage of	000438
Iodine 129 and Technetium 99*	000004
Iodine 129 in Nuclear Fuel Reprocessing and Waste	000285
Iodine 129*	000062
Ion Exchange Decontamination of Alkaline Solutions of	000486
Ion Exchange Equilibria*	000676
Ion Exchange Properties of a Representative Oil Sand*	000029
Ion Exchange Research by the United States Geological	000068
Ion Exchange Resin Beads*	000670
Ions in Unsaturated Soils after Water Disposal into a	000292
Ions*	000006
Irradiated Fuel Cladding*	000416
Irradiation on Migration of Brine Inclusions in a Salt	000081
Irradiation* Damage Studies on Natural and Synthetic	000082
Island, Alaska* Distribution in a Cavity and Chimney	000377
Isolated in Geologic Formations*	000166
Isolation in Geologic Formations in the United States	000528
Isolation in Geologic Formations. Baseline Rock	000021
Isolation in Geologic Formations. Facility Construction	000586
Isolation in Geologic Formations. Stratigraphics of	000385
Isolation in Geologic Formations. Volume 10. Repository	000587
Isolation in Geologic Formations. Volume 12. Repository	000588
Isolation in Geologic Formations. Volume 14. Repository	000589
Isolation in Geologic Formations. Volume 16. Repository	000550
Isolation in Geologic Formations. Volume 19 - Thermal	000075
Isolation in Geologic Formations. Volume 20.	000024
Isolation in Geologic Formations. Volume 21. Ground	000237
Isolation in Geologic Formations. Volume 23 -	000296
Isolation in Geologic Formations. Volume 4. Baseline	000384
Isolation in Geologic Formations. Volume 6. Baseline	000022
Isolation in Geologic Formations. Volume 7. Baseline	000023
Isolation in Geologic Formations. Volume 8. Repository	000585
Isolation in Geologic Formations* Impact Statement	000457
Isolation in Rock-Forming Minerals*	000149
Isolation of Nuclear Waste* @Program Scenario Analysis	000587
Isolation of Radioactive Waste* @The Status	000536
Isolation of Solid Radioactive Waste in Fine-Grained	000376
Isolation Canisters*	000102
Isolation Facilities*	000553
Isolation Facility Design*Salt Migration to Dissolution	000263
Isolation Facility*	000554
Isolation Implications*	000220
Isolation Pilot Plant (WIPP) Rocks* Pilot Plant (WIPP)	000073
Isolation Pilot Plant (WIPP) Site in New Mexico* of	000241
Isolation Pilot Plant (WIPP) Site, Southeastern New	000073
Isolation Pilot Plant (WIPP) Site, Southeastern New	000074
Isolation Pilot Plant (WIPP) Site, Southeastern New	00034
Isolation Pilot Plant (WIPP) Site, Southeastern New	000453
Isolation Pilot Plant (WIPP) Site, Southeastern New	000454
Isolation Pilot Plant (WIPP) Site, Southeastern New	000455
Isolation Pilot Plant (WIPP) Site, Southeastern New	000456
Isolation Pilot Plant (WIPP). December 15, 1978-March	000007
Isolation Pilot Plant (WIPP). October 1-December 15,	000008
Isolation Pilot Plant (WIPP), March 15-June 15, 1979*	000009
Isolation Pilot Plant (WIPP), October 1-December 15,	000007
Isolation Pilot Plant - A Status Report*	000482
Isolation Pilot Plant in New Mexico* of Nuclides	000300
Isolation Pilot Plant Facility*	000494
Isolation Pilot Plant In Situ Experimental Program for	000137
Isolation Pilot Plant Mine Drift*	000532
Isolation Pilot Plant Site, Calendar Year 1976*	000363
Isolation Pilot Plant*	000537
Isolation Pilot Plant* Inclusions in Permian Salado	000072
Isolation Pilot Project (WIPP), April 1-June 30, 1978*	000160
Isolation Program Annual Report*	000451
Isolation Program Monthly Report, April 1978*	000390
Isolation Program Monthly Report, February 1978*	000387
Isolation Program Monthly Report, January 1978*	000388
Isolation Program Monthly Report, March 1978*	000389
Isolation Projects: Fiscal Year 1978*	000447
Isolation Repositories*	000369
Isolation Repository* @Material	000120

PERMUTED TITLE INDEX

Control and Accountability Considerations for a Waste
Berkeley Laboratory Contract No. 45901AK. Waste
Source Term Characterization Task at @Overview of Waste
Waste Canister Corrosion Studies Pertinent to Geologic
in Preparing Dense Cements for Radioactive Waste
on Cementitious Materials for Nuclear Waste
Containment of Radioactive Waste* @Hot
Granite: Results and Preliminary @Geochemistry and
Commercially Operated Low-Level @Evaluation of
Commercially Low-Level Radioactive Waste @Evaluation of
Commercially Operated Low-Level @Evaluation of
Services, Incorporated, Reprocessing Plant at West @
@Disposal of Long-Lived Radioactive Wastes in
@Petrology of Tuff Units from the
Petrology of Tuff Units from the J-13 Drill Site,
@Statement of Dr.
@Pacific Northwest Laboratory Monthly Activities Report
@Basalt Waste Isolation Program Monthly Report,
River Laboratory Quarterly Report, Waste Management,
Division Fuel Cycle Programs Progress Report,
Liquid Transuranic Waste from the Nuclear Fuel Cycle:
@Radioactive Waste Management in
Used in Shale Fracturing Grouts as Determined by the
Atlantic Coastal Plain, North Carolina through New
Shale Fracturing Grouts as Determined by the Jenike and
@Deformation of a
Level @Non-Radioactive Operation Experience with a
@Sedimentary History and Economic Geology of San
-- Natural Fission Reactor Program, Progress Report for
Report, Process Technology and Process Development,
and Reprocessing Division of the Department of Energy (
System for U.S. Department of Energy Defense Waste,
@Waste Management Programs Quarterly Report,
Liquid Transuranic Waste from the Nuclear Fuel Cycle:
River Laboratory Quarterly Report, Waste Management,
Waste Cyclone Drum Incinerator and Treatment System:
in the Waste Isolation Pilot Plant (WIPP), March 15-
Transportation Quarterly Progress Report, April Through
National Waste Terminal Storage Program Progress Report
Waste Isolation Pilot Project (WIPP), April 1-
Problem* @Dissolution of
@Subsurface Waste-Disposal Potential in Salina Basin of
@Salt Dissolution in Oil and Gas Test Holes in
Lincoln, Ellsworth, Barton, and Rice Counties, Central
@Salt Dissolution in Oil and Gas Test Holes in Central
of Radioactive Wastes at the Nuclear Research Center
Potential at the Proposed Waste Isolation Pilot @Batch
@Canada
*
@Radioactive Waste Management in the United
Repository Site Characterization* @An Evaluation of
Evaluation for the Continuous Monitoring of Carbon 14,
of Process and Storage Materials Suitable for
Studies of the Degradation of RAD-Wastes for the Sandia
Source Term Characterization Task at Pacific Northwest
@Present Environment of the Idaho National Engineering
Selected Rocks: 1977 Annual Report of Lawrence Berkeley
@Pacific Northwest
@
March 1975* @Savannah River
September, 1975* @Savannah River
@Field and
of Alternatives for Idaho National Engineering
Term Management of Stored Idaho National Engineering
Design Studies: Battelle Pacific Northwest
Disposal by Shale Fracturing at Oak Ridge National
Disposal by Shale Fracturing at Oak Ridge National
Transuranic Waste at the Idaho National Engineering
and Capabilities within the Los Alamos Scientific
Research and Development Program at Oak Ridge National
for Alpha Bearing Wastes at the Los Alamos Scientific
and Disposal on the Idaho National Engineering
Long Range Plan, Idaho National Engineering
Transuranic Waste at the Idaho National Engineering
Management Complex at the Idaho National Engineering
240 Concentrations in a Stream near Argonne National
@Suggested Concentration Limits for Shallow
Radioactive Waste @Evaluation of Isotope Migration-
Low-Level Radioactive @Evaluation of Isotope Migration-
Low-Level Radioactive @Evaluation of Isotope Migration-
@The Disposal of Radioactive Waste on
@Statement of Dr. Terry R.
Wastes in the Federal Republic of Germany. Vol 8. The
Transport in Selected Rocks: 1977 Annual Report of
Induced Optical @Characterization of Leached Surface
@Design Aspects of the Alpha Repository: 2. Conceptual
Site of a Potential Nuclear Waste Repository, Eddy and
Chemical Decontamination and Melt Densification of Chop-
Isolation Repository* @An Overview of Material 000496
Isolation Safety Assessment Program - Collection and 000219
Isolation Safety Assessment Program and Description of 000315
Isolation* @High-Level 000103
Isolation* @Role of Admixtures 000662
Isolation* @Permeability Measurements 000156
Isostatic Pressing for the Consolidation and 000629
Isotope Hydrology of Groundwaters in the Stripa 000397
Isotope Migration-Land Burial Water Chemistry at 000166
Isotope Migration-Land Burial Water Chemistry at 000167
Isotope Migration-Land Burial Water Chemistry at 000168
Issues Related to the Closing of the Nuclear Fuel 000169
Italy* 000403
J-13 Drill Site, Jackass Flats, Nevada* 000819
Jackass Flats, Nevada* 000810
James E. Martin* 000518
January 1978* 000216
January 1978* 000388
January-March 1975* @Savannah 000668
January-March 1978* @Chemical Engineering 000236
January-March 1978* Reduction System for Solid and 000570
Japan* 000509
Jenike and Johanson Method* @Flowability of Bulk Solids 000681
Jersey* Aquifer System Below the Zone of Fresh Water, 000366
Johanson Method* @Flowability of Bulk Solids Used in 000681
Jointed Rock Mass* 000866
Joule Heated Ceramic Melter for Vitrification of High- 000686
Juan Basin, New Mexico and Colorado* 000483
July 1-September 30, 1979* @Ohio 000065
July 1977 through September 1977* Operations Quarterly 000335
July-December 1976)* for Waste Management, Production, 000490
July-December 1977* @Fluidized Bed Incineration 000389
July-September 1978* 000672
July-September 1978* Reduction System for Solid and 000571
July-September, 1975* @Savannah 000159
July-September, 1978* @TRU 000622
June 15, 1979* Wastes Under Conditions Expected 000009
June 1976* @Nuclear Waste Management and 000333
June 1977* @ 000312
June 30, 1978*of RAD-Wastes for the Sandia Laboratories 000160
Kansas Evaporites - The Radioactive Waste Disposal 000859
Kansas* 000398
Kansas* 000478
Kansas* Part 1: Salt Beds in the Subsurface in Russell, 000479
Kansas, Part 1: Salt Beds in the Subsurface in Russell, 000479
Karlsruhe - Experience from Plant Operation and 000612
Kd Measurements of Nuclides to Estimate Migration 000300
Keeps its Disposal Options Open* 000473
Kinetic Model for Silicate Rock-Ground Water Interaction 000088
Kingdom* 000612
Kriging Techniques for High-Level Radioactive Waste 000026
Krypton 85 and Iodine 129 in Nuclear Fuel Reprocessing 000285
Krypton 85 Waste Management* @The Development 000181
Laboratories Waste Isolation Pilot Project (WIPP), 000160
Laboratories* Assessment Program and Description of 000315
Laboratory (Including RWMC, Site 14, and Lemhi Range)* 000207
Laboratory Contract No. 45901AK. Waste Isolation Safety 000219
Laboratory Monthly Activities Report January 1978* 000216
Laboratory Pillar Model Experiments* 000051
Laboratory Quarterly Report, Waste Management, January- 000668
Laboratory Quarterly Report, Waste Management, July- 000199
Laboratory Studies Preliminary to Main Experiment* 000012
Laboratory Transuranic Waste Management*@Identification 000875
Laboratory Transuranic Waste* of Alternatives for Long- 000280
Laboratory Waste Forms in Salt*Repository Preconceptual 000550
Laboratory* @Waste 000213
Laboratory* @Waste 000216
Laboratory* @Retrieval of Buried 000184
Laboratory* @Waste Management Programs 000171
Laboratory* @High-Level Waste Management 000570
Laboratory* @Waste Management Capabilities 000819
Laboratory* Involving Retrieval, Processing, 000382
Laboratory* @Decontamination and Decommissioning, 000603
Laboratory* and Interim Storage of Intermediate-Level 000187
Laboratory* Review Document for the Radioactive Waste 001176
Laboratory* @A Preliminary Assessment of Plutonium 239, 000201
Land Burial of Radionuclides* 000277
Land Burial Water Chemistry at Commercially Low-Level 000167
Land Burial Water Chemistry at Commercially Operated 000166
Land Burial Water Chemistry at Commercially Operated 000168
Land* 000871
Leak* 000511
Law Concerning the Disposal of Radioactive Wastes* 000895
Lawrence Berkeley Laboratory Contract No. 45901AK. 000219
Layers on Simulated High-Level Waste Glasses by Sputter- 000118
Layouts of Underground Storage Facilities* 000501
Lea Counties, Mexico* of Hydrologic Conditions Near the 000831
Leach Fuel Hulls* @ 000588

PERMUTED TITLE INDEX

Forms* $ Leach Rate Characterization of Solid Radioactive Waste 000286
of Radioactive Waste Solution with Zeolite (2). Leachabilities of Cesium and Strontium from Calcined 000619
$Procedure for Determining Leachabilities of Radioactive Waste Forms* 000088
Containing Savannah River $Correlation of Radionuclide Leachabilities with Microstructures of Glasses 000620
$Microstructures and Leachability of Vitrified Radioactive Wastes* 000659
on Amounts of Cesium 137, Strontium 90 and Cobalt 60 Leached from Matrix of Hardened Cement Grout* Specimen 000629
Glasses by Sputter-Induced Optical $Characterization of Leached Surface Layers on Simulated High-Level Waste 000118
$ Leaching of Fully Radioactive High-Level Waste Glass* 000358
Blocks Buried Below the Water Table: Fifteen Years $The Leaching of Radioactivity from Highly Radioactive Glass 000685
$Application of Mass Transport Theory to the Leaching of Radionuclides from Waste Solids* 000258
$Long-Term Elevated Temperature Leaching of Solid Waste Forms* 000128
$Calculations of Radioactivity Release Due to Leaching of Vitrified High Level Waste* 000591
Reactor Waste Glass* $ Leaching Characteristics of Actinides from Simulated 000153
$241-T-106 Tank Leak Investigation* 000162
$ Leak Testing Fuel Stored in the ICPP Fuel Storage Basin* 000631
$High-Level Waste Leakage from the 241-T-106 Tank at Hanford* 000198
Storage Basin Water* $Primary Source of Radionuclides Leaking into the Idaho Chemical Processing Plant Fuel 000188
$Alternatives Involving Leave-in-Place Management* 000210
Engineering Laboratory (Including BWEC, Site 14, and Lemhi Range)*$Present Environment of the Idaho National 000207
Solved Waste Problems* $ Let's Go Slowly on a Nuclear Power Program Until We've 000527
Repositories* $State/Federal Licensing Interactions for Radioactive Waste 000584
$Function and Licensing Status of the Asse Test Storage Facility* 000887
$Neutron and Gamma-Ray Sources in Light Water Reactor High-Level Nuclear Waste* 000590
Disposal of Non-Fuel Bearing Hardware Resulting from Light Water Reactor Operations* Transportation, and 000651
Package Loss at Sea* $Evaluation of the Consequences of Light Water Reactor Spent Fuel and Plutonium Shipping 000118
"As Low As Practicable" Guides - Fabrication of Light-Water Reactor Fuel from Enriched Uranium Dioxide* 000652
$The Alternatives for Managing Post- Light-Water Reactor Nuclear Wastes* 000519
Formations* $ Light-Water Reactor Wastes to be Isolated in Geologic 000186
Properties* $Sources of Radioactive Waste from Light-Water Reactors and Physical and Chemical 000691
$Report of Investigation Underground Limestone Mines in the Ohio Region* 000370
Creek Clay, and Yazoo Clay in Parts of $Basal Ottawa Limestone, Chattanooga Shale, Floyd Shale, Porters 000430
Ions in Unsaturated Soils after Water Disposal into a Limited Surface, Part 2* Soils: Behavior of Radioactive 000292
$Suggested Concentration Limits for Shallow Land Burial of Radionuclides* 000277
Kansas, Part 1: Salt Beds in the Subsurface in Russell, Lincoln, Ellsworth, Barton, and Rice Counties, Central 000479
Cations* $ Linde 13X Equilibrium with Trivalent Cerium and Yttrium 000092
$Operation of the Monticello Nuclear Generating Plant Liquid and Solid Radioactive Waste Disposal System* 000545
$The Movement of Liquid through Soils* 000265
$Subsurface Disposal of Liquid Industrial Wastes by Deep-Well Injection* 000880
of Organic Coolants for the Transportation of Liquid Metal Fast Breeder Reactor Spent Fuel Rods* 000096
$Influence of Radiation on the System Liquid Radioactive Wastes - Geologic Formation* 000078
Basis for, and Experience With, Underground Storage of Liquid Radioactive Wastes in the USSR* $The Scientific 000468
Thin-Film Evaporators* $Volume Reduction of Liquid Radioactive Wastes Using Mechanically Agitated 000592
for Using Deep Water-Bearing Horizons for Disposal of Liquid Radioactive Wastes* Prerequisites and Practice 000203
January-March $Volume Reduction System for Solid and Liquid Transuranic Waste from the Nuclear Fuel Cycle: 000570
July-September $Volume Reduction System for Solid and Liquid Transuranic Waste from the Nuclear Fuel Cycle: 000571
December 1977* $Volume Reduction System for Solid and Liquid TRU Waste from the Nuclear Fuel Cycle: October- 000638
- Development, Part #4 Model for Calculating Effects of Liquid Waste Disposal in Deep Saline Aquifers, Part 1 - 000267
$Radiological Problems of Liquid Waste Disposal into the Sea* 000281
of Actinide-Bearing Sediments Underlying Liquid Waste Disposal Facilities at Hanford* 000280
$ Liquid Waste Evaporation and Acid Recovery* 000666
Heated Ceramic Melter for Vitrification of High-Level Liquid Waste* Operation Experience with a Joule 000686
$Radioactive Waste Disposal Report on Seepage Pit Liquid Waste-Shale Column Experiment* 000178
of Strontium and Cesium in Intermediate-Level Liquid Wastes by Solidification in Cements* 000668
$Potential of Denver Basin for Disposal of Liquid Wastes* 000401
and Storage of Hanford's High-Level Radioactive Liquid Wastes* $Solidification 000669
for Removal of Radioactive Waste (Sludge, Saltcake and Liquid)*Well Access Using Underground Mining Techniques 000630
$Ultimate Storage of High-Level Waste Solids and Liquids in Salt Formations* 000359
$Ultimate Storage of High-Level Waste Solids and Liquids in Salt Formations* 000360
$Guide to Radioactive Waste Management Literature* 000323
$Trace-Element Transport in Lithic Material by Fluid at High Temperatures* 000297
$The Migration of Long- Lived Radioactive Processing Wastes in Selected Rocks* 000250
$High-Level and Long- Lived Radioactive Waste Disposal* 000351
$Disposal of Long- Lived Radioactive Wastes in Italy* 000403
$Geochemical Behavior of Long- Lived Radioactive Wastes* 000258
$Fire Protection Device for LMFBR Spent Fuel Shipping Casks* 000568
of Waste-Container Sleeve: 1. Initial Estimates of Loading on the Sleeve* $Thermoelastic/Plastic Analysis 000067
Stress Analysis and Development of Thermal Loading Guidelines* Volume 20. Thermomechanical 000028
Sleeve: 3. Influence of Salt Strength on Sleeve Loading* Plastic Analysis of Waste-Container 000080
$Geologic and Hydrologic Factors for Locating Repositories for Radioactive Wastes* 000427
$Importance of Deep Permeable Disposal Formations in Location of a Large Nuclear-Fuel Reprocessing Plant* 000391
Waste and Spent Fuel* $ Logistics Models for the Transportation of Radioactive 000566
$Waste Management Programs and Capabilities within the Los Alamos Scientific Laboratory* 000171
Management Capabilities for Alpha Bearing Wastes at the Los Alamos Scientific Laboratory* $Waste 000819
Water Reactor Spent Fuel and Plutonium Shipping Package Loss at Sea* $Evaluation of the Consequences of Light 000118
$Thermal Conductivity of Rock Salt from Louisiana Salt Domes* 000061
Relative to the National Waste Terminal Storage $As Low as is Reasonably Achievable (ALARA) Studies 000493
in the Nuclear Fuel Cycle for Use in Establishing "As Low As Practicable" Guides - Fabrication of Light-Water 000652
Media* $Steady Thermal Convection at Low Rayleigh Number from Concentrated Sources in Porous 000037
Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites* Migration- 000166
Land Burial Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites* Migration- 000168
Migration-Land Burial Water Chemistry at Commercially Low-Level Radioactive Waste Disposal Sites* of Isotope 000167
$Operation of Low-Level Radioactive Waste Incinerator* 000588
* $A Transport Model for Long Term Release of Low-Level Radionuclide Solutions into a Stream Ecosystem 000311
Scientific Basis for Physical and Chemical Testing* $ Low-Level Solidified Waste Characterization: The 000003
$Radionuclide Migration from Low-Level Waste: A Generic Overview* 000302
Impact Program (ARIP), Part I. Carcinogenic Hazard from Low-Level, Low-Rate Radiation*$The Argonne Radiological 000253
(ARIP), Part I. Carcinogenic Hazard from Low-Level, Low-Rate Radiation* Argonne Radiological Impact Program 000253
Used and Products Formed with Radioactive Wastes at LWR Nuclear Power Plants* of Solidification Agents 000621

PERMUTED TITLE INDEX

of Relative Areas for the Disposal in Bedded Salt of *Geology and Description of the
Plutonium from Waste Streams* *Evaluation of Dev *Estimate
*Field and Laboratory Studies Preliminary to
*Critical Pathways of Radionuclides to
Technological Program for Radioactive Waste
Wastes* *The
Report, October Through December 1975* *Nuclear Waste
Report, April Through June 1976* *Nuclear Waste
*Radioactive Waste
*High-Level Nuclear Waste
*Radioactive Waste
*Waste
Plants* *Long-Term Planning for
*The
*
Idaho Chemical Processing *Alternatives for Long-Term
Hanford Reservations, *Alternatives for Long-Term
*Summary - Alternatives for Long-Term
River Plant, Aiken, South *Alternatives for Long-Term
Wastes from a Fuel Reprocessing Plant* *
Power Plants* *
*
and Other Evaluations of Alternatives for Long-Term
*Effects of Current Transuranic Waste
and 2: Progress Report for Period April 1 to *Waste
of Significant Elements and Radionuclides for Waste
Los Alamos Scientific Laboratory* *Waste
*The Radioactive Waste
*Safety Review Document for the Radioactive Waste
*Systems Analysis Study for Waste
*Radioactive Waste
*Guide to Radioactive Waste
*National Waste Terminal Storage Program, Configuration
*Development of Advanced Waste
Station* *High-Level Radioactive Waste
*The High-Level Radioactive Waste
Alamos Scientific Laboratory* *Waste
1976* *Waste
Pacific* *High-Level Radioactive Waste
Ridge National Laboratory* *High-Level Waste
Program, Configuration Management Plan. Volume 1.
*International Waste-
*Alternative Waste
of a Risk Assessment Method for Radioactive Waste
Nuclear Fuel Cycle with Emphasis on High-Level Waste
*Nuclear Waste
*Alternatives Involving Leave-in-Place
*Scientific Basis for Nuclear Waste
*Nuclear Waste
*High-Level Radioactive Waste
Regulatory Commission Program for High Level Waste
Waste Management--Review of Commercial Nuclear Waste
of Natural Retention Experience to Nuclear Waste
Effects from Current Operations of Transuranic Waste
Accidental Radioactive Release in Nuclear Waste
and Storage Materials Suitable for Krypton 85 Waste
Environmental Risks Associated with Radioactive Waste
Idaho National Engineering Laboratory Transuranic Waste
of Separation Methods Applied to Radioactive Waste
Agency Program for Reprocessing and High Level Waste
Management* *Nuclear Waste
*Savannah River Laboratory Quarterly Report, Waste
*Savannah River Laboratory Quarterly Report, Waste
Report, Waste Incineration and Fixation for Waste
Environmental Impact Statement on Commercial Waste
*Science Underlying Radioactive Waste
*Radioactive Waste
*The Alternatives for
*Environmental Effects of Alternatives for
Fuels Preliminary Description and Operating Procedures
*Deformation
Expected in the Waste Isolation Pilot Plant (WIPP),
Waste Isolation Pilot Plant (WIPP). December 15, 1978-
Laboratory Quarterly Report, Waste Management, January-
*Basalt Waste Isolation Program Monthly Report,
Division Fuel Cycle Programs Progress Report, January-
Transuranic Waste from the Nuclear Fuel Cycle: January-
Green Alga (Ulva pertusa)* *Influences of
*Statement of Dr. James E.
of the Volatility of Cesium from Aluminosilicates by
Implementation (Discrete-Parcel-Random- *Multicomponent
from Waste Solids* *Application of
*Deformation of a Jointed Rock
Constraints on Accumulation of Actinide Critical
Suitability for Possible Storage of Radioactive Waste
*Trace-Element Transport in Lithic

LWR Wastes from Alternative Fuel Cycles* *Estimates 000050
Lyons Site* 000328
Macroporous Resins for the Removal of Uranium and 000623
Magnitude of Waste Disposal Problems* 000698
Main Experiment* 000012
Fate from Agro-Ecosystems* 000255
Management and Reprocessing* Development Administration 000325
Management and Storage of Commercial Power Reactor 000681
Management and Transportation Quarterly Progress 000226
Management and Transportation Quarterly Progress 000322
Management in the United Kingdom* 000613
Management in the United States: A Time for Decisions* 000513
Management in Japan* 000509
Management of Actinide Contaminated Soil* 000648
Management of Aqueous Wastes from Fuel Reprocessing 000523
Management of Canada's Nuclear Wastes* 000317
Management of Commercial High-Level Radioactive Waste* 000551
Management of Defense High-Level Radioactive Waste 000346
Management of Defense High-Level Radioactive Waste 000347
Management of Defense High-Level Radioactive Waste* 000476
Management of Defense Transuranic Waste at the Savannah 000206
Management of Highly Active Decladding Zircaloy Solid 000097
Management of Nontritium Radioactive Wastes from Fusion 000701
Management of Radioactive Wastes* 000327
Management of Stored Idaho National Engineering 000340
Management on the Workers and on the Public* 000209
Management Analysis for the Nuclear Fuel Cycle-Parts 1 000687
Management Assessment* *Selection 000703
Management Capabilities for Alpha Bearing Wastes at the 000419
Management Complex and Its Operations* 0208
Management Complex at the Idaho National Engineering 0176
Management Criteria* 0 505
Management Development in Europe* 000684
Management Literature* 000323
Management Plan. Volume 1. Management Summary* 000499
Management Plans* 000225
Management Program at the National Reactor Testing 000671
Management Program at Nuclear Fuel Services* 000589
Management Programs and Capabilities within the Los 000171
Management Programs Quarterly Report, July-September 000672
Management Research and Development Program at Battelle 000644
Management Research and Development Program at Oak 000579
Management Summary* *National Waste Terminal Storage 000499
Management Symposium* 000522
Management Systems* 000318
Management. Volume 2. Implementation for Terminal 000279
Management. Volume 2*An Analysis of the Back End of the 000321
Management* 000190
Management* 000210
Management* 000320
Management* 000524
Management* 000582
Management* *Nuclear 000508
Management* *Nuclear 000610
Management* *Relevance 000584
Management* *Environmental 000307
Management* *Methodology for Estimating 000278
Management* *The Development of Process 000181
Management* *Risk Assessment Methodology of 000259
Management* *Identification of Alternatives for 000475
Management* *Technology Assessment and Impact Analysis 000601
Management*Description of the Environmental Protection 000328
Management--Review of Commercial Nuclear Waste 000610
Management, January-March 1975* 000668
Management, July-September, 1975* 000199
Management, Production, and Reprocessing Division of 000490
Management: Radioactive Waste Isolation in Geologic 000457
Management: Status and Needs* 000663
Management: The Need for Multiple Barriers* 000150
Managing Post-Light-Water Reactor Nuclear Wastes* 000519
Managing Transuranic Waste* 000308
Manual* Processing Plant Waste Canisters and Spent 000138
Mapping of Polycrystalline Salt* 000010
March 15-June 15, 1979* Wastes Under Conditions 000009
March 15, 1979* Wastes Under Conditions Expected in the 000007
March 1975* *Savannah River 000668
March 1978* 000389
March 1978* *Chemical Engineering 000336
March 1978*Volume Reduction System for Solid and Liquid 000570
Marine Sediment on the Accumulation of Radionuclides by 000285
Martin* 000514
Mass Spectrometry* *Investigations 000138
Mass Transport Model: Theory and Numerical 000215
Mass Transport Theory to the Leaching of Radionuclides 000258
Mass* 000466
Masses from Stored Nuclear Waste in Natural Rock 000015
Material as of September, 1977.* to Investigate their 000439
Material by Fluid at High Temperatures* 000297

PERMUTED TITLE INDEX

a Waste Isolation Repository* ©An Overview of
Waste Isolation Repository* ©
Element Galerkin Model* ©
Suitability for Possible Storage of Radioactive Waste
of Sealing Boreholes with Compacted Natural Earthen
©Permeability Measurements on Cementitious
the Relative ©Survey of Naturally Occurring Hazardous
©Behavior of Candidate Canister
©Transportation of Radioactive Waste-
©The Development of Process and Storage
Gas Generation in Plutonium Contaminated Waste
of Sealing Boreholes with Compacted Natural Earthen
©
Heterogeneous Clayey Formation ©Application to the Case ©
Transport ©Annual Report October 1977 to September 1978
©Evaluation of Concrete as a
of Cesium 137, Strontium 90 and Cobalt 60 Leached from
Containers During Retrievable Geologic Storage* ©
Crust* ©An Analysis of the
Solidification* ©
©An Automatic Drum Scanning System for the
©Thermal Effects in Shales:
Report on Geophysical and Mechanical Borehole
Potential at the Proposed Waste Isolation ©Batch KS
Nuclide Migration* ©
Waste Isolation* ©Permeability
Studies on Natural and Synthetic Rock Salt Utilizing
Area 17, Nevada Test Site, Using the Transient Method* ©
Radioactive Waste Repositories in Hard Rock. Parts 1 ©
Compression up to 200 C* ©
©Preliminary Report on Geophysical and
©The
©Electrical Heaters for Thermo-
©Volume Reduction of Liquid Radioactive Wastes Using
©Analysis and Evaluation of the Rock
©Summary Review of Rock
©Potential
©Microstructural Interactions of Geologic
Mobility ©Radionuclide Interaction with Soil and Rock
©Nuclide Migration Models for Faulted
©Dispersion in Flow through Porous
©Flow in Fractured Porous
©Sorption and Migration of Radionuclides in Geologic
©Transport Properties of Nuclear Waste in Geologic
©Fixation of Radioactivity in Stable, Solid
©Fixation of Radioactivity in Stable, Solid
Salt Approach to Nuclear Waste Disposal in Geologic
Low Rayleigh Number from Concentrated Sources in Porous
of High Active Waste with Crystalline Rock Repository
During Adsorption of Multivalent Elements by Geologic
©Calculations of Nuclide Migration in Rock and Porous
©Material Transport through Porous
©Studies of Nuclear-Waste Migration in Geologic
into Bitumen* ©Recent Experiments on the Treatment of
Suitability of Anhydrite as a Nuclear Waste Repository
©National Waste Terminal Storage Review
Media* ©The Rock-
©Chemical Decontamination and
Operation Experience with a Joule Heated Ceramic
of Nuclear Waste Disposal by Underground In Situ
at Waste-Storage Potential in Selected Segments of the
Factors Related to the Waste Storage Potential of
©
of Organic Coolants for the Transportation of Liquid
©Studies on the Hydrolysis Reactions of
©Bonding of
©Corrosion of
Radioactive Waste Disposal* ©
Iodine 129 and Technetium 99* ©
Releases in Nuclear Waste Management* ©
Radioactive Waste Management* ©An Assessment
History and Economic Geology of San Juan Basin, New
©Mechanical Behavior of New
©New
Isolation Pilot Plant (WIPP) Site, Southeastern New
Isolation Pilot Plant (WIPP) Site, Southeastern New
Isolation Pilot Plant (WIPP) Site, Southeastern New
Isolation Pilot Plant (WIPP) Site, Southeastern New
Isolation Pilot Plant (WIPP) Site, Southeastern New
Isolation Pilot Plant (WIPP) Site, Southeastern New
Studies Associated with the WIPP Site in Southern New
Nuclear Waste Repository, Eddy and Lea Counties,
with the Waste Isolation Pilot Plant (WIPP) Site in New
Isolation Pilot Plant (WIPP) Site, Southeastern New
at the Proposed Waste Isolation Pilot Plant in New
Studies of Fluid Inclusions in Permian Salado Salt, New
©Geologic Description of Cores from Holes P-3
Descriptions of Cores from Holes P-3 WH-1 through P-3
Material Control and Accountability Considerations for
Material Control and Accountability Procedures for a
Material Transport through Porous Media: A Finite-
Material* Texas Salt Dome Basin to Investigate Their
Material, Volumes 1 and 2* ©Feasibility
Materials for Nuclear Waste Isolation*
Materials in Deep Geologic Formations: A Perspective on
Materials in Deep Ocean Environments*
Materials to the Sun*
Materials Suitable for Krypton 85 Waste Management*
Materials* ©Radiolytic
Materials, Volume 3* ©Feasibility
Mathematical Model of Waste Storage Operations*
Mathematical Modelling of Radionuclide Migration in a
Mathematical Simulation of Sediment and Contaminant
Matrix for Solidification of Savannah River Plant Waste*
Matrix of Hardened Cement Grout* of Specimen on Amounts
Maximum Acceptable Temperatures of Wastes and
Measured Values for the State of Stress in the Earth's
Measurement and Control of Cement Set Times in Waste
Measurement of Plutonium in Waste*
Measurements and Modeling*
Measurements at Strips* ©Preliminary
Measurements of Nuclides to Estimate Migration
Measurements of Parameters Essential to Predicting
Measurements on Cementitious Materials for Nuclear
Measurements Made During Electron Irradiation* Damage
Measuring the Permeability of Kleges Argillite from
Mechanical and Thermal Design Considerations for
Mechanical Behavior of New Mexico Rock Salt in Triaxial
Mechanical Borehole Measurements at Strips*
Mechanical Properties of Strips Granite*
Mechanical Tests at the Strips Mine*
Mechanically Agitated Thin-Film Evaporators*
Mechanic= Aspects of the Proposed Salt Mine Repository*
Mechanic= Workshop on Radioactive Waste Disposal*
Mechanisms of Containment Failure*
Media with Waste Radionuclides*
Media. Volume 1: Processes Influencing Radionuclide
Media*
Media*
Media*
Media*
Media*
Media*
Media*
Media* ©The Rock-
Media* ©Steady Thermal Convection at
Media* ©Long Term Geochemical Interactions
Media* ©Electron Transfer Method of Controlling Eh
Media, Penetrated by Water*
Media: A Finite-Element Galerkin Model*
Media: Annual Report, November 1976-October 1977*
Medium Level Wastes and Spent Solvent and on Fixation
Medium* and Brine and Their Implications for the
Meeting on Salt Dissolution*
Melt Approach to Nuclear Waste Disposal in Geologic
Melt Densification of Chop-Leach Fuel Hulls*
Melter for Vitrification of High-Level Liquid Waste*
Melting* ©Economic and Environmental Evaluation
Mesozoic Aquifer System Below the Zone of Fresh Water,
Mesozoic Aquifers in the Southern Part of the Atlantic
Metal Encapsulation of Ceramic Nuclear*
Metal Fast Breeder Reactor Spent Fuel Rods* ©Evaluation
Metal Ions*
Metal Oxides in Sodium Silicate Glass*
Metals in a Hot, High Pressure Brine Environment*
Metasiot Mineral Alteration: An Implication for
Methodology for the Determination of Environmental
Methodology for Estimating Accidental Radioactive
Methodology of Environmental Risks Associated with
Mexico and Colorado* ©Sedimentary
Mexico Rock Salt in Triaxial Compression up to 200 C*
Mexico Waste Isolation Pilot Plant - A Status Report*
Mexico. Geochemistry* Characterization Report, Waste
Mexico. Hydrology* Characterization Report, Waste
Mexico. Regional Geology* Characterization Report, Waste
Mexico. Seismology* Characterization Report, Waste
Mexico. Site Geology* Characterization Report, Waste
Mexico. Special Studies of Waste Isolation Pilot Plant (
Mexico* ©Radionuclide Migration
Mexico* Conditions Near the Site of a Potential
Mexico* of Radionuclides with Geomedia Associated
Mexico* ©Geological Characterization Report, Waste
Mexico* of Nuclides to Estimate Migration Potential
Mexico, to Problems of Siting the Waste Isolation Pilot
WH-1 through P-3 EH-5, Area 6, Technical Area 54*
WH-5, Area 6, Technical Area 54* ©Geologic

PERMUTED TITLE INDEX

®Geology of Subsurface Waste Disposal in	Michigan Basin® 000364
®Applicability of	Microautoradiography to Sorption Studies® 000003
Expected in the Waste Isolation Pilot Plant (®Potential	Microbial Impact on Transuranic Wastes Under Conditions 000007
Expected in the Waste Isolation Pilot Plant (®Potential	Microbial Impact on Transuranic Wastes Under Conditions 000008
Expected in the Waste Isolation Pilot Plant (®Potential	Microbial Impact on Transuranic Wastes Under Conditions 000009
®Potential	Microbial Impact on Transuranic Wastes® 000228
Transuranic Elements in Soils, ®The Relationship of	Microbial Processes to the Fate and Behavior of 000310
System Resulting in an Extremely ®Synthetic Rutile	Microencapsulation: A Radioactive Waste Solidification 000197
®The Sorption of	Microquantities of Strontium and Cesium in Soils® 000045
Simulated Reactor Wasteforms® 0	Microstructural Characterization of Solidified 000115
Waste Radionuclides® 0	Microstructural Interactions of Geologic Media with 000038
Radioactive Wastes® 0	Microstructures and Leachability of Vitrified 000659
Plant ®Correlation of Radionuclide Leachabilities with	Microstructures of Glasses Containing Savannah River 000620
for Developing Dry Underground Storage Space in the	Midcontinent, U.S.A.® Search for Carbonates Suitable 000371
®Radionuclide	Migration from Low-Level Waste: A Generic Overview® 000302
Application to ®Mathematical Modelling of Radionuclide	Migration in a Homogeneous Clayey Formation (000291
of Strontium 90 Migration® ®Radionuclides	Migration in Aerated Zones. 3. Theoretical Estimation 000288
®In-Situ Experiments on Nuclide	Migration in Fractured Crystalline Rocks® 000276
1976-October 1977® ®Studies of Nuclear-Waste	Migration in Geologic Media: Annual Report, November 000299
®Calculations of Nuclide	Migration in Rock and Porous Media, Saturated by Water® 000262
Temperature Gradients, Stress, and Irradiation on	Migration of Brine Inclusions in a Salt Repository® 000041
in Selected Rocks® ®The	Migration of Long-Lived Radioactive Processing Wastes 000250
®A Study of the	Migration of Radioactive Elements in Soils "In-Place"® 000220
®Sorption and	Migration of Radionuclides in Geologic Media® 000288
Examples® ®Thermodynamic Considerations Underlying the	Migration of Radionuclides in Geomedia: Oklo and Other 000231
Nevada Test Site - Part ®Information Pertinent to the	Migration of Radionuclides in Ground Water at the 000163
Different Texture® 0	Migration of Strontium 90 and Cesium 137 in Soils of 000227
232-Thorium 232 Cycle® 0	Migration of Uranium Process Wastes from the Uranium 000251
Affects a Radioactive Waste ®The Response of Salt	Migration to Dissolution in Salt Domes and How It 000262
®Nuclide	Migration Models for Vaulted Media® 000232
Applications to the Problem of Geological Storage of 0	Migration Paths for Oklo Reactor Products and 000236
Pilot ®Batch Kd Measurements of Nuclides to Estimate	Migration Potential at the Proposed Waste Isolation 000309
®Geologic	Migration Potentials of Technetium 99 and Neptunium 237® 000011
Southern New Mexico® ®Radionuclide	Migration Studies Associated with the WIPP Site in 000280
®The Effect of Retardation Factors on Radionuclide	Migration® 000257
®Geochemistry and Radionuclide	Migration® 000269
of Parameters Essential to Predicting Nuclide	Migration® ®Measurements 000280
Zones. 3. Theoretical Estimation of Strontium 90	Migration® ®Radionuclides Migration in Aerated 000288
Operated Low-Level Radioactive ®Evaluation of Isotope	Migration-Land Burial Water Chemistry at Commercially 000166
Low-Level Radioactive Waste ®Evaluation of Isotope	Migration-Land Burial Water Chemistry at Commercially 000167
Operated Low-Level Radioactive ®Evaluation of Isotope	Migration-Land Burial Water Chemistry at Commercially 000168
®Forced-Air Cooling of a Waste Isolation Pilot Plant	Mine Drift® 000532
Cavity--Computer Model and Calibration from Detroit	Mine Experiment® ®Growth and Shape of Salt 000462
0	Mine Renovation and Preparation of Experimental Area® 000395
of the Rock Mechanics Aspects of the Proposed Salt	Mine Repository® ®Analysis and Evaluation 000178
®Geology and Description of the Lyons	Mine® 000328
®Borehole Drilling and Related Activities at the Stripa	Mine® 000422
Heaters for Thermo-Mechanical Tests at the Stripa	Mine® ®Electrical 000538
on the Disposal of Radioactive Wastes in the Asse Salt	Mine® ®Recent Results and Developments 000177
Program, Bleasa Investigation® 0	Mineability Study of the Terminal Waste Storage 000396
Cooperative Program on Radioactive Waste Storage in	Mined Caverns in Crystalline Rock, Program Summary® 000407
®Earth Science Technical Plan for	Mined Geologic Disposal of Radioactive Waste® 000503
Waste Disposal® ®Retsyict	Mineral Alteration: An Implication for Radioactive 000030
0	Mineral Exchange at the University of North Carolina® 000677
®Hanford	Mineral Exchange Program® 000669
0	Mineral Reaction Work at Hanford® 000572
®PIRN Wastes: Permanent Isolation in Rock-Forming	Minerals® 000189
®Sorption and Retention by Clay	Minerals: Oak Ridge Studies® 000675
®Survey of Active and Inactive	Mines for Possible Use as In Situ Test Facilities® 000378
®Report of Investigation Underground Limestone	Mines in the Ohio Region® 000370
®Hydrology of Some Deep	Mines in Precambrian Rocks® 000489
®Thermal Conductivity of the Rocks in the Bureau of	Mines Standard Rock Suite® 000063
of High Activity Solidified Wastes in Underground Salt	Mines® Salt Vault: A Demonstration of the Disposal 000316
Wastes® ®Operational Techniques to	Minimize, Segregate, and Account for Alpha-Bearing 000618
Rooms at Stripa® 0	Mining Methods Used in the Underground Tunnels and Test 000750
Retrieval System-Tank Wall Access Using Underground	Mining Techniques for Removal of Radioactive Waste (000630
Structural Analysis of Salt Cavities Formed by Solution	Mining: 1. Method of Analysis and Preliminary Results 000540
0	Miscellaneous Radioactive and Nonradioactive Wastes® 000695
Porters Creek Clay, and Tuxedo Clay in Parts of Alabama,	Mississippi and Tennessee as Potential Host Rocks for 000630
®Fixation of Radioactive Wastes in Clay-Flux	Mines® 000682
for Self Fixation of Radioactive Wastes in Clay-Flux	Mixtures® ®Heat Experiments 000617
Media. Volume 1: Processes Influencing Radionuclide	Mobility and Retention; Element Chemistry and 000217
®Growth and Shape of Salt Cavity--Computer	Model and Calibration from Detroit Mine Experiment® 000462
in Deep Saline Aquifers. Part 1 — Development, Part ®A	Model for Calculating Effects of Liquid Waste Disposal 000267
Solutions into a Stream Ecosystem® ®A Transport	Model for Long Term Release of Low-Level Radionuclide 000311
®Kinetic	Model for Silicate Rock-Ground Water Interaction® 000080
Report on Development of a Semi-Empirical Numerical	Model for Simulating the Deformation Behavior of a High- 000020
®Mathematical	Model of Waste Storage Operations® 000667
®Illite: A	Model Based on the Sorption Behavior of Cesium® 000033
®Laboratory Pillar	Model Experiments® 000051
of Some Tracer Dans in Granite Rock Using a Pressure	Model® ®Analysis 000287
through Porous Media: A Finite-Element Galerkin	Model® ®Material Transport 000243
Parcel-Random-Walk ®Multicomponent Mass Transport	Model: Theory and Numerical Implementation (Discrete- 000215
and Ocean Disposal Concepts Regarding Realistic	Modeling that Allows Objective Risk Assessment to be 000517
®Thermal Effects in Shales: Measurements and	Modeling® 000057
Clayey Formation (Application to the Case ®Mathematical	Modelling of Radionuclide Migration in a Homogeneous 000291
Options® 0	Modelling Studies Used to Evaluate Waste Disposal 000260
Spent Fuel® ®Logistics	Models for the Transportation of Radioactive Waste and 000566

PERMUTED TITLE INDEX

@Nuclide Migration	Models for Faulted Media* 000232
@Development of High-Level Waste Shipping Cask	Models* 000112
@Comparisons Among Alternatives, Concepts and	Modulus* 000145
Formation (Application to the Case of the Boom Clay at	Mobility of Radionuclide Migration in a Homogeneous Clayey 000291
of Nuclear Waste by In-Situ Incorporation in Deep	Molten Silicate Rock* @Disposal 000632
@Synthesis of Nuclear Waste	Monazites, Ideal Actinide Hosts for Geologic Disposal* 000640
@A Continuous Tritium	Monitor for Nuclear Facility Off-Gas Streams* 000172
Calendar Year 1976* @Environmental	Monitoring at the Waste Isolation Pilot Plant Site, 000363
Nuclear Fuel @Methods Evaluation for the Continuous	Monitoring of Carbon 14, Krypton 85 and Iodine 129 in 000265
@Corrosion	Monitoring of Storage Bins for Radioactive Calcines* 000116
Fuel @Methods Evaluation and Development for the	Monitoring of Technetium 99 and Selenium 79 in Nuclear 000596
* @Selection of a Criticality	Monitoring Technique for a Transuranic Waste Incinerator 000650
@Environmental	Monitoring* 000555
@Clay-Phosphate Ceramics and Nitrates: Alternatives to	Monolithic High Level Waste Products* 000089
@Pacific Northwest Laboratory	Monthly Activities Report January 1976* 000216
@Basalt Waste Isolation Program	Monthly Report, April 1978* 000398
@Basalt Waste Isolation Program	Monthly Report, February 1978* 000397
@Basalt Waste Isolation Program	Monthly Report, January 1978* 000388
@Basalt Waste Isolation Program	Monthly Report, March 1978* 000389
Radioactive Waste Disposal System* @Operation of the	Monticello Nuclear Generating Plant Liquid and Solid 000595
Sodium-Cesium* @	Montmorillonite Exchange Equilibria with Strontium- 000027
@Salt	Motion Following Nuclear Waste Disposal* 000386
@Nuclear Waste Canister Thermally Induced	Motion* 000025
@Plutonium Distribution in a Desert Pavement-Desert	Mound Soil System in Area 11* 000305
@Thermal Properties of Cable	Mountain Basalt Cores Hanford Nuclear Reservation* 000052
of Talc and Serpentinite Bodies from the Appalachian	Mountain Region of Eastern United States with Regard to 000488
in Geologic Formations. Volume 2D. Ground Water	Movement and Nuclide Transport* @Waste Isolation 000237
@Accumulation and	Movement of Fission Products in Soils and Plants* 000058
@The	Movement of Liquid through Soils* 000265
@	Movement of Radiostrontium in Soils* 000306
@Development of	Multibarrier Nuclear Waste Forms* 000665
Numerical Implementation (Discrete-Parcel-Random-Walk @	Multicomponent Mass Transport Model: Theory and 000215
@Radioactive Waste Management: The Need for	Multiple Barriers* 000150
Transfer Method of Controlling Eh During Adsorption of	Multivalent Elements by Geologic Media* @Electron 000070
Research on Waste Disposal Off the Coast of Tokai-	Mura* @Oceanographic 000015
as a Function of pH and Certain Soil Components: NMJ-	NaOH System*238 and Curium 242 from a Contaminated Soil 000285
1A, and Lochi Mangoji* @Present Environment of the Idaho	National Engineering Laboratory (Including INEL Site 000207
Management* @Identification of Alternatives for Idaho	National Engineering Laboratory Transuranic Waste 000075
Alternatives for Long-Term Management of Stored Idaho	National Engineering Laboratory Transuranic Waste* of 000340
@Retrieval of Buried Transuranic Waste at the Idaho	National Engineering Laboratory* 000188
and Decontamination, Long Range Plan, Idaho	National Engineering Laboratory* @Decontamination 000403
of Intermediate-Level Transuranic Waste at the Idaho	National Engineering Laboratory* and Interim Storage 000187
the Radioactive Waste Management Complex at the Idaho	National Engineering Laboratory* Review Document for 000176
Retrieval, Processing, and Disposal on the Idaho	National Engineering Laboratory*@Alternatives Involving 000382
@Waste Disposal by Shale Fracturing at Oak Ridge	National Laboratory* 000212
@Waste Disposal by Shale Fracturing at Oak Ridge	National Laboratory* 000218
Research and Development Program at Oak Ridge	National Laboratory* @High-Level Waste Management 000570
239, 240 Concentrations in a Stream near Argonne	National Laboratory*Preliminary Assessment of Plutonium 000201
@Operating Practices, Experiences, and Problems at the	National Reactor Testing Station* 000200
@High-Level Radioactive Waste Management Program at the	National Reactor Testing Station* 000671
Report* @Hydrology of Waste Disposal	National Reactor Testing Station, Idaho, An Interim 000418
June 1977* @	National Waste Terminal Storage Program Progress Report 000312
for Period October 1, 1976, to September 30, 1977* @	National Waste Terminal Storage Program Progress Report 000313
Criteria* @Development of	National Waste Terminal Storage Program Waste Packaging 000127
@Preliminary Geologic Site Selection Factors For the	National Waste Terminal Storage Program* 000488
Reasonably Achievable (ALARA) Studies Relative to the	National Waste Terminal Storage Program* @As Low as is 000493
Management Plan. Volume 1. Management Summary* @	National Waste Terminal Storage Program, Configuration 000490
Fractions in the Waste Transportation System* @	National Waste Terminal Storage Program: Potential 000568
Dissolution* @	National Waste Terminal Storage Review Meeting on Salt 000388
Applications* @Radiation Damage Studies on	Natural and Synthetic Rock Salt for Waste Disposal 000081
Made During Electron @Radiation Damage Studies on	Natural and Synthetic Rock Salt Utilizing Measurements 000087
@Feasibility of Sealing Boreholes with Compacted	Natural Earthen Material, Volumes 1 and 2* 000052
@Feasibility of Sealing Boreholes with Compacted	Natural Earthen Materials, Volume 3* 000066
July 1-September 30, 1979* @Ohio --	Natural Fission Reactor Program, Progress Report for 000065
@	Natural Glasses: Analogues for Radioactive Waste Forms* 000111
Ground of @Conclusions Drawn from Studies on the Ohio	Natural Nuclear Reactor Concerning the Evolution in the 000287
* @Relevance of	Natural Retention Experience to Nuclear Waste Management 000588
Actinide Critical Masses from Stored Nuclear Waste in	Natural Rock Repositories* on Accumulation of 000017
Interaction of a Glass-Based Nuclear Waste Simulant and	Natural Rock* @ 000080
Geologic Formations: A Perspective on the @Survey of	Naturally Occurring Hazardous Materials in Deep 000088
@Oceanic Research	Needed for Safe Disposal of Radioactive Wastes at Sea* 000480
@Chemistry of Technetium 99 and	Neptunium 237 in Contact with Unweathered Igneous Rocks* 000228
@Geologic Migration Potentials of Technetium 99 and	Neptunium 237* 000011
Differing Weathering @Sorption of Technetium 99,	Neptunium 237, and Americium 241 on Two Subsoils from 000288
States Subsoils from Differing @Americium 241,	Neptunium 237, and Technetium 99 Sorption on Two United 000293
of Distribution Ratios and Diffusion Coefficients of	Neptunium, Americium, and Curium in Soil-Aquatic 000301
@	Net Energy from Nuclear Power* 000520
High-Level Nuclear Waste* @	Neutron and Gamma-Ray Sources in Light Water Reactor 000590
Chemical Processing Plant Waste Canisters and Spent @	Neutron Interrogator Assay System for the Idaho 000138
River Valley Salt Deposits, Clark County, Southeastern	Nevada to Investigate their Suitability for Possible 000439
the Migration of Radionuclides in Ground Water at the	Nevada Test Site - Part 2: Annotated Bibliography* to 000163
@Water Experiments in the Climax Stock,	Nevada Test Site* 000485
the Permeability of Bloom Argillite from Area 17,	Nevada Test Site, Using the Transient Method*Measuring 000388
of Tuff Units from the J-13 Drill Site, Jackass Flats,	Nevada* @Petrology 000810
Annual Report of Lawrence Berkeley Laboratory Contract	No. 0590188. Waste Isolation Safety Assessment Program 000210
General @Engineered Storage for the Disposal of	Non-Fuel Bearing Components from Nuclear Facilities and 000094
Reduction, Packaging, Transportation, and Disposal of	Non-Fuel Bearing Hardware Resulting from Light Water 000651
Heated Ceramic Melter for Vitrification of High-Level @	Non-Radioactive Operation Experience with a Joule 000686

PERMUTED TITLE INDEX

@Miscellaneous Radioactive and	Nonradioactive Wastes*	000695
@Management of	Nutritive Radioactive Wastes from Fusion Power Plants*	000701
Below the Zone of Fresh Water, Atlantic Coastal Plain,	North Carolina through New Jersey* Aquifer System	000366
@Mineral Exchange at the University of	North Carolina*	000677
Disposal of Industrial Wastes in Subsurface Rocks on	North Flank of Appalachian Basin in New York* for	000829
on the Disposal of Radioactive Wastes into the	North Pacific* @Evaluation	000303
@Geologic Study of the Interior Salt Domes of	Northeast Texas Salt Dome Basin to Investigate Their	000439
@Regional Geology of the	Northeast Texas Salt Dome Basin*	000412
@Genesis and Growth of Salt Domes in the	Northeast Texas Salt Dome Basin*	000432
@Geology of Certain Domes of the	Northeast Texas Salt-Dome Basin*	000353
of Source Term Characterization Task at Pacific	Northwest Laboratories* Program and Description	000315
1974* @Pacific	Northwest Laboratory Monthly Activities Report January	000216
Preconceptual Design Studies: Battelle Pacific	Northwest Laboratory Waste Forms in Salt*16. Repository	000550
Waste Migration in Geologic Media: Annual Report,	November 1976-October 1977* @Studies of Nuclear-	000299
@Environmental Geology in Development of	Nuclear Energy*	000052
in a Cavity and Chimney Produced by the Cannikin	Nuclear Explosion, Amchitka Island, Alaska*Distribution	000377
@Waste Incineration and Immobilization for	Nuclear Facilities (April-September 1977)*	000616
@Decommissioning of	Nuclear Facilities - An Annotated Bibliography*	000326
for the Disposal of Non-Fuel Bearing Components from	Nuclear Facilities and General Radioactive Waste*	000094
@A Continuous Tritium Monitor for	Nuclear Facility Off-Gas Streams*	000172
Reprocessing Waste. 4. Safety @Handling of Spent	Nuclear Fuel and Final Storage of Vitrified High-Level	000123
@Shipments of	Nuclear Fuel and Waste...Are They Really Safe?*	000152
and the Environmental Impact of Waste Effluents in the	Nuclear Fuel Cycle - Conversion of Yellow Cake to	000521
and the Environmental Impact of Waste Effluents in the	Nuclear Fuel Cycle - Reprocessing of High-Temperature	000501
and the Environmental Impact of Waste Effluents in the	Nuclear Fuel Cycle for Use in Establishing "As Low As	000652
Management. Volume 20An Analysis of the Back End of the	Nuclear Fuel Cycle with Emphasis on High-Level Waste	000321
United States and in Doar from an Integrated	Nuclear Fuel Cycle Facility* Doar from the Southeastern	000271
Period April 1 to @Waste Management Analysis for the	Nuclear Fuel Cycle-Parts 1 and 2: Progress Report for	000647
System for Solid and Liquid Transuranic Waste from the	Nuclear Fuel Cycle: January-March 1978*Volume Reduction	000570
System for Solid and Liquid Transuranic Waste from the	Nuclear Fuel Cycle: July-September 1978* Reduction	000571
System for Solid and Liquid TRU Waste from the	Nuclear Fuel Cycle: October-December 1977* Reduction	000634
for the Monitoring of Technetium 99 and Selenium 79 in	Nuclear Fuel Reprocessing and Waste Solidification	000598
Monitoring of Carbon 14, Krypton 85 and Iodine 129 in	Nuclear Fuel Reprocessing and Waste Solidification Off-	000265
@Radioactive Waste Discharges to the Environment from a	Nuclear Fuel Reprocessing Plant*	000633
@The High-Level Radioactive Waste Management Program at	Nuclear Fuel Services*	000589
at West Valley, @Issues Related to the Closing of the	Nuclear Fuel Services, Incorporated, Reprocessing Plant	000169
Resistance of Canisters for Final Disposal of Spent	Nuclear Fuel* @Corrosion	000132
Waste Disposal System* @Operation of the Monticello	Nuclear Generating Plant Liquid and Solid Radioactive	000595
@Disposal of Radioactive Wastes Produced in	Nuclear Installations in the German Democratic Republic*	000050
@Retrofitting Trojan	Nuclear Plant's Spent Basin Transfer System*	000658
@	Nuclear Power in Sweden*	000322
@Radioactive Waste Discharges to the Environment from	Nuclear Power Facilities*	000280
of Spent Fuel to be Discharged by the United States	Nuclear Power Industry* @Projections	000690
@Summary of Radioactivity Released in Effluents from	Nuclear Power Plants from 1973 thru 1976*	000708
@Incorporation of Radioactive Wastes from	Nuclear Power Plants into Concrete and Bitumen*	000653
used and Products Formed with Radioactive Wastes at LWR	Nuclear Power Plants* of Solidification Agents	000621
@Let's Go Slowly on a	Nuclear Power Program Until We've Solved Waste Problems*	000527
@Handling of Radioactive Waste at Soviet	Nuclear Power Stations*	000624
@Net Energy from	Nuclear Power*	000520
of @Conclusions Drawn from Studies on the Oklo Natural	Nuclear Reactor Concerning the Evolution in the Ground	000247
@Carbon 14 Production in	Nuclear Reactors*	000698
Waste Management* @	Nuclear Regulatory Commission Program for High Level	000508
@Risk Assessment Review Group Report to the U.S.	Nuclear Regulatory Commission*	000512
Plant @Bituminization of Radioactive Wastes at the	Nuclear Research Center Karlsruhe - Experience from	000612
Properties of Gable Mountain Basalt Cores Hanford	Nuclear Reservation* @Thermal	000053
@	Nuclear Safety and the Environment*	000332
Comment* @Western New York	Nuclear Service Center Study, Final Report for Public	000211
of Soil Samples from Test Trench at Western New York	Nuclear Service Center, West Valley, New York* Analysis	000173
@Description of Allied-General	Nuclear Services On-Site Solid Waste Storage Concepts*	000205
@	Nuclear Shipping and Waste Disposal Cost Estimates*	000587
@Interactions Between	Nuclear Waste and Surrounding Rock*	000282
Silicate Rock* @Disposal of	Nuclear Waste by In-Site Incorporation in Deep Holes	000632
@The Disposal of High-Level	Nuclear Waste in the Oceans*	000477
@Transport Properties of	Nuclear Waste in Geologic Media*	000290
on Accumulation of Actinide Critical Masses from Stored	Nuclear Waste in Natural Rock Repositories* Constraints	000015
@	Nuclear Waste Canister Thermally Induced Motion*	000025
Corrosion Test of a Glass-Ceramic Candidate for a	Nuclear Waste Canister* @Preliminary	000151
@Economic and Environmental Evaluation of	Nuclear Waste Disposal by Underground In Situ Melting*	000498
@The Rock-Melt Approach to	Nuclear Waste Disposal in Geologic Media*	000017
@Salt Motion Following	Nuclear Waste Disposal*	000386
@Are Plutons the Answer to	Nuclear Waste Disposal?*	000318
@	Nuclear Waste Disposal: The Geological Aspects*	000375
Interactions* @Hydrothermal Reactivity of Simulated	Nuclear Waste Forms and Water-Catalyzed Waste-Rock	000058
@Characterization of Glass and Glass Ceramic	Nuclear Waste Forms*	000635
@Development of Multibarrier	Nuclear Waste Forms*	000665
of Prediction* @Chemistry of	Nuclear Waste Glass Reactions: Problems and Potential	000093
@Devitrification Behavior in a Zinc Borosilicate	Nuclear Waste Glass*	000679
@Corrosion Considerations for	Nuclear Waste Isolation Canisters*	000102
Permeability Measurements on Cementitious Materials for	Nuclear Waste Isolation* @	000156
Progress Report, October Through December 1975* @	Nuclear Waste Management and Transportation Quarterly	000226
Progress Report, April Through June 1976* @	Nuclear Waste Management and Transportation Quarterly	000333
for Decisions* @High-Level	Nuclear Waste Management in the United States: A Time	000513
@	Nuclear Waste Management*	000190
@Scientific Basis for	Nuclear Waste Management*	000330
@	Nuclear Waste Management*	000528
@Relevance of Natural Retention Experience to	Nuclear Waste Management*	000584
@Nuclear Waste Management--Review of Commercial	Nuclear Waste Management*	000610
for Estimating Accidental Radioactive Releases in	Nuclear Waste Management* @Methodology	000278

PERMUTED TITLE INDEX

Waste Management*	0 Nuclear Waste Management--Review of Commercial Nuclear	000010
Geologic Disposal*	0Synthesis of Nuclear Waste Monoliths, Ideal Actinide Waste for	000040
0Gold and Selective Storage Salves	Nuclear Waste Problem*	000159
Implications for the Suitability of Anhydrite as a	Nuclear Waste Repository Radius* and Brine and Their	000016
0Computer Enhanced "Release Scenario" Analysis for a	Nuclear Waste Repository*	000060
of Hydrologic Conditions Near the Site of a Potential	Nuclear Waste Repository, Eddy and Lea Counties, Mexico*	000031
0Interaction of a Glass-Based	Nuclear Waste Simulant and Natural Rock*	000040
of a Titanate Based Ceramic for High Level	Nuclear Waste Solidification* 0Characterization	000122
0Feasibility of Space Disposal of Radioactive	Nuclear Waste. 1. Executive Summary*	000038
0Feasibility of Space Disposal of Radioactive	Nuclear Waste. 2. Technical Summary*	000033
0Rock Stresses Associated with Burial of	Nuclear Waste*	000085
0Underground Disposal of Canada's	Nuclear Waste*	000070
0Ion Exchange Decontamination of Alkaline Solutions of	Nuclear Waste*	000086
and Gamma-Ray Sources in Light Water Reactor High-Level	Nuclear Waste* 0Neutron	000590
in Assessing the Safety of the Geologic Isolation of	Nuclear Waste*Program Scenario Analysis Methods for Use	000582
Impacts*	0 Nuclear Waste: Increasing Scal. and Sociopolitical	000510
	0 Nuclear Waste: Twenty-Five Yr a Place ... and Beyond*	000372
	0 Nuclear Wastes in the Deep Seabed*	000956
0The Management of Canada's	Nuclear Wastes*	000117
0The Alternatives for Managing Post-Light-Water Reactor	Nuclear Wastes*	000519
0Radiolytic Gas Production during Long-Term Storage of	Nuclear Wastes*	000577
0U.S. Program for the Immobilization of High-Level	Nuclear Wastes*	000506
Applications to the Problem of Geological Storage of	Nuclear Wastes* Paths for Ohio Reactor Products and	000236
A Perspective on the Relative Hazard of Deep Burial of	Nuclear Wastes* Materials in Deep Geologic Formations:	000088
0Oceanic Distribution of Radionuclides from	Nuclear Weapons Testing*	000230
0Total Encapsulation of Ceramic	Nuclear*	000618
Permeable Disposal Formations in Location of a Large	Nuclear-Fuel Reprocessing Plant* 0Importance of Deep	000391
Report, November 1976-October 1977* 0Studies of	Nuclear-Waste Migration in Geologic Media: Annual	000299
0In-Situ Experiments on	Nuclide Migration in Fractured Crystalline Rocks*	000276
by Water* 0Calculations of	Nuclide Migration in Rock and Porous Media, Penetrated	000262
	0 Nuclide Migration Models for Faulted Media*	000232
0Measurements of Parameters Essential to Predicting	Nuclide Migration*	000240
	0 Nuclide Transport by Groundwater in Swedish Bedrock*	000001
Formations. Volume 21. Ground Water Movement and	Nuclide Transport* Waste Isolation in Geologic	000237
0Effects of Radiations from Transuranium	Nuclides on Container Surfaces*	000133
Proposed Waste Isolation 0Batch Kd Measurements of	Nuclides to Estimate Migration Potential at the	000300
0Multicomponent Mass Transport Model: Theory and	Numerical Implementation (Discrete-Parcel-Random-Walk	000215
of a 0Interim Report on Development of a Semi-Empirical	Numerical Model for Simulating the Deformation Behavior	000020
and Brine Invasion into Fresh Water Aquifer. 3.	Numerical Simulation of Brine Invasion of Fresh Water	000275
and Brine Invasion into Fresh Water Aquifer. 1.	Numerical Study of Convective-Diffusive Flow in	000273
	0 NWTS In-Situ Test Beds Workshop, August 29-30, 1978*	000383
0Waste Disposal by Shale Fracturing at	Oak Ridge National Laboratory*	000113
0Waste Disposal by Shale Fracturing at	Oak Ridge National Laboratory*	000218
Waste Management Research and Development Program at	Oak Ridge National Laboratory* 0High-Level	000579
0Sorption and Retention by Clay Minerals:	Oak Ridge Studies*	000675
0A Review of the Near Surface Heater Experiment at	Oak Ridge, Tennessee*	000021
Concepts Regarding Realistic Modeling that Allows	Objective Risk Assessment to be Made*and Ocean Disposal	000517
0In Situ Heating Experiments in Hard Rock: Their	Objectives and Design*	000382
Disposal* 0Pollucite and Its Alteration on Geological	Occurrences and in Deep-Burial Radioactive Waste	000105
that Allows Objective 0Comparison of Some Geologic and	Ocean Disposal Concepts Regarding Realistic Modeling	000517
0Behavior of Candidate Canister Materials in Deep	Ocean Environments*	000188
Aspects of Disposal of High-Level Waste on the	Ocean Floor* 0Assessment of the Radiological Protection	000261
the Possibility of Disposing of Radioactive Waste in	Ocean Trenches* 0Concerning	000357
	0 Ocean Waste Disposal (A Bibliography with Abstracts)*	000367
0A Bibliography on	Ocean Waste Disposal*	000465
Weapons Testing*	0 Oceanic Distribution of Radionuclides from Nuclear	000230
Radioactive Wastes at Sea*	0 Oceanic Research Needed for Safe Disposal of	000489
of Total-Sure*	0 Oceanographic Research on Waste Disposal Off the Coast	000415
0The Disposal of High-Level Nuclear Waste in the	Oceans*	000477
and Transportation Quarterly Progress Report,	October Through December 1975*0Nuclear Waste Management	000226
Expected in the Waste Isolation Pilot Plant (WIPP).	October 1-December 15, 1978* Wastes Under Conditions	000068
Expected in the Waste Isolation Pilot Plant (WIPP),	October 1-December 15, 1978* Storage Under Conditions	000067
Terminal Storage Program Progress Report for Period	October 1, 1976, to September 30, 1977* 0National Waste	000313
Waste Repository Sites: Progress Report for the Period	October 1, 1977, to September 30, 1978* in Radioactive	000059
of Sediment and Contaminant Transport in 0Annual Report	October 1977 to September 1978 Mathematical Simulation	000289
in Geologic Media: Annual Report, November 1976-	October 1977* 0Studies of Nuclear-Waste Migration	000299
Solid and Liquid TRU Waste from the Nuclear Fuel Cycle:	October-December 1977* 0Volume Reduction System for	000634
of Investigation Underground Limestone Mines in the	Ohio Region* 0Report	000370
Beds in the Subsurface in Russell, 0Salt Dissolution in	Oil and Gas Test Holes in Central Kansas, Part 1: Salt	000479
0Salt Dissolution in	Oil and Gas Test Holes in Kansas*	000478
0Ion Exchange Properties of a Representative	Oil Sand*	000029
Salt Deposits in the Texas Panhandle and Western	Oklahoma for Underground Storage of Radioactive Wastes*	000417
	0 Oklo - Geologic Isolation Implications*	000220
Report for July 1-September 30, 1979*	0 Oklo -- Natural Fission Reactor Program, Progress	000065
Underlying the Migration of Radionuclides in Geomedia:	Oklo and Other Examples* 0Thermodynamic Considerations	000231
in the Ground of 0Conclusions Drawn from Studies on the	Oklo Natural Nuclear Reactor Concerning the Evolution	000287
of Geological Storage of Nuclear 0Migration Paths for	Oklo Reactor Products and Applications to the Problem	000236
0Canada Ponders Its Disposal Options	Open*	000873
Migration-Land Burial Water Chemistry at Commercially	Operated Low-Level Radioactive Waste Disposal Sites*	000166
Migration-Land Burial Water Chemistry at Commercially	Operated Low-Level Radioactive Waste Disposal Sites*	000168
	0 Operating Experience*	000109
	0 Operating Practices and Experiences at Hanford*	000179
National Reactor Testing Station*	0 Operating Practices, Experiences, and Problems at the	000200
Canisters and Spent Fuels Preliminary Description and	Operating Procedures Manual* Processing Plant Waste	000138
Research Center Karlsruhe - Experience from Plant	Operation and Development Work* Wastes at the Nuclear	000612
Liquid and Solid Radioactive Waste Disposal System*	0 Operation of the Monticello Nuclear Generating Plant	000595
	0 Operation of Low-Level Radioactive Waste Incinerator*	000588
for Vitrification of High-Level Liquid 0Non-Radioactive	Operation Experience with a Joule Heated Ceramic Melter	000686

PERMUTED TITLE INDEX

Account for Alpha-Bearing Wastes* *
*Environmental Effects from Current
Process Development, July 1977 *Rockwell Hanford
*The Radioactive Waste Management Complex and Its
*Mathematical Model of Waste Storage
Bearing Hardware Resulting from Light Water Reactor
Simulated High-Level Waste Glasses by Sputter-Induced
*Canada Keep: Its Disposal
*Modelling Studies Used to Evaluate Waste Disposal
Fast Breeder Reactor Spent Fuel Rods* *Evaluation of
*Retention of
Countries Pursue Geological Disposal* *
Stage of Transporting Solid *Experience in the
Fortuna Creek Clay, and Yazoo Clay in Parts of *Basal
Considerations for a Waste Isolation Repository* *An
and Description of Source Term Characterization Task *
*Radionuclide Migration from Low-Level Waste: A Generic
*Bonding of Metal
*Confinement of Volatilized Ruthenium
Implications for the Suitability of Anhydrite as a *
*Geologic Description of Cores from Holes
Description of Cores from Holes P-3 WH-1 through
and Description of Source Term Characterization Task at
January 1978* *
96. Repository Preconceptual Design Studies: Battelle
on the Disposal of Radioactive Wastes into the Bedrock
Management Research and Development Program at Battelle
*Radioactive Waste
Light Water Reactor Spent Fuel and Plutonium Shipping
*Sea Disposal of
of National Waste Terminal Storage Program Waste
*Review of Criteria for
*Spent Fuel Handling and
Bearing Hardware *Experience in Volume Reduction,
Indiana* *
of *Evaluation of Permian Salt Deposits in the Texas
*Measurements of
Model: Theory and Numerical Implementation (Discrete-
*Incentives for
Problem of Geological Storage of Nuclear *Migration
*Critical
*Plutonium Distribution in a Bonnet
Processing Plant Zirconia Wastes* *Development of a
of Brine Migration in Rock and Porous Media,
*Plutonium-Americium Soil
Subsurface Waste Disposal in a Structural Syncline in
*TITE Vistas:
Test Site, Using the Transient Method* *Measuring the
*Stress, Strain, and Temperature Induced
Nuclear Waste Isolation* *
Pressure of Granitic Rocks* *Experimental
*Compressible Fluid Flow Through Rocks of Variable
Nuclear-Fuel Reprocessing Plant* *Importance of Deep
the *Application of Studies of Fluid Inclusions in
Western Oklahoma for Underground Storage *Evaluation of
Hazardous Materials in Deep Geologic Formations: A
of High-Level Radioactive Wastes - Earth-Science
Water at the Nevada Test Site - Part 2: *Information
*High-Level Waste Canister Corrosion Studies
*Summary of Previous
the Accumulation of Radionuclides by Green Alga (Ulva
Jackson Flats, Nevada* *
Cerium 242 from a Contaminated Soil as a Function of
Waste in Glass. Part 2. The Experimental Evaluation of
Acmolithic High Level Waste Products* *Clay-
of Radioactive Waste from Light-Water Reactors and
Waste Characterization: The Scientific Basis for
Temperature and Pressure to *Application of
*United States Geological Survey Studies of
Agents used and Products Formed with Radioactive *
*Laboratory
*A
New Mexico. Special Studies of Waste Isolation
with Gases Associated with the Waste Isolation
*Geological Characterization Report, Waste Isolation
*Geological Characterization Report, Waste Isolation
*Geological Characterization Report, Waste Isolation
*Geological Characterization Report, Waste Isolation
*Geological Characterization Report, Waste Isolation
*Geological Characterization Report, Waste Isolation
*Geological Characterization Report, Waste Isolation
Wastes Under Conditions Expected in the Waste Isolation
Wastes Under Conditions Expected in the Waste Isolation
Wastes Under Conditions Expected in the Waste Isolation
Under Conditions Expected in the Waste Isolation
*New Mexico Waste Isolation
Migration Potential at the Proposed Waste Isolation
*Preliminary Safety Assessment of the Waste Isolation

Operational Techniques to Minimize, Segregate, and
Operations of Transuranic Waste Management*
Operations Quarterly Report, Process Technology and
Operations*
Operations*
Operations* Transportation, and Disposal of Non-Fuel
Optical Emission* of Leached Surface Layers on
Options Open*
Options*
Organic Coolants for the Transportation of Liquid Metal
Organic-Complexed Plutonium in Soil*
Organization for Economic Cooperation and Development
Organization of Sanitary Radiometric Control at the
Ottawa Limestone, Chattanooga Shale, Floyd Shale,
Overview of Material Control and Accountability
Overview of Waste Isolation Safety Assessment Program
Overview*
Oxides in Sodium Silicate Glass*
Oxides*
P-T-X Relations of Anhydrite and Brine and Their
P-3 WH-1 through P-3 WH-5, Area G, Technical Area 54*
P-3 WH-7, Area G, Technical Area 54* *Geologic
Pacific Northwest Laboratories* Assessment Program
Pacific Northwest Laboratory Monthly Activities Report
Pacific Northwest Laboratory Waste Forms in Salt*Volume
Pacific* *Evaluation
Pacific* *High-Level Radioactive Waste
Package Development at the Rocky Flats Plant*
Package Loss at Sea* *Evaluation of the Consequences of
Packaged Radioactive Wastes*
Packaging Criteria* *Development
Packaging Plutonium for Transport by Air*
Packaging Program. A Survey of Hot Cell Facilities*
Packaging, Transportation, and Disposal of Non-Fuel
Paleozoic Stratigraphy of Two Areas in Southeastern
Panhandle and Western Oklahoma for Underground Storage
Parameters Essential to Predicting Nuclide Migration*
Parcel-Random-Walk Version)* Mass Transport
Partitioning High-Level Waste*
Paths for Ohio Reactor Products and Applications to the
Pathways of Radionuclides to Man from Agro-Ecosystems*
Pavement-Desert Round Soil System in Area 11*
Pelleted Waste Form for High-Level Idaho Chemical
Penetrated by Water* *Calculations
Penetration at 234-5 Building Crib Sites*
Pennsylvania* *Possibilities for
Permanent Isolation in Rock-Forming Minerals*
Permeability of Eleana Argillite from Area 17, Nevada
Permeability Changes in Potential Repository Rocks*
Permeability Measurements on Cementitious Materials for
Permeability Studies at Elevated Temperature and
Permeability*
Permeable Disposal Formations in Location of a Large
Permian Salado Salt, New Mexico, to Problems of Siting
Permian Salt Deposits in the Texas Panhandle and
Perspective on the Relative Hazard of Deep Burial of
Perspectives* *Geologic Disposal
Pertinent to the Migration of Radionuclides in Ground
Pertinent to Geologic Isolation*
Pertinent Studies*
pertuna)* *Influences of Marine Sediment on
Petrology of Tuff Units from the J-13 Drill Site,
pH and Certain Soil Components: NaNO3-NaOH System* and
Phosphate and Borosilicate Glasses* of Radioactive
Phosphate Ceramics and Vitromets: Alternatives to
Physical and Chemical Properties* *Sources
Physical and Chemical Testing* *Low-Level Solidified
Physical Chemistry of Fluids in Rock Salt at Elevated
Physical-Chemical Relationships in Salt Repositories*
Physicochemical Characterization of Solidification
Pillar Model Experiments*
Pilot Heater Test in the Strips Granite*
Pilot Plant (WIPP) Rocks* (WIPP) Site, Southeastern
Pilot Plant (WIPP) Site in New Mexico* of Radionuclides
Pilot Plant (WIPP) Site, Southeastern New Mexico.
Pilot Plant (WIPP) Site, Southeastern New Mexico.
Pilot Plant (WIPP) Site, Southeastern New Mexico.
Pilot Plant (WIPP) Site, Southeastern New Mexico.
Pilot Plant (WIPP) Site, Southeastern New Mexico.
Pilot Plant (WIPP) Site, Southeastern New Mexico. Site
Pilot Plant (WIPP) Site, Southeastern New Mexico*
Pilot Plant (WIPP). December 15, 1978-March 15, 1979*
Pilot Plant (WIPP). October 1-December 15, 1978*
Pilot Plant (WIPP), March 15-June 15, 1979* Transuranic
Pilot Plant (WIPP), October 1-December 15, 1978*Storage
Pilot Plant - A Status Report*
Pilot Plant in New Mexico* of Facilities to Estimate
Pilot Plant Facility*

REPORTED TITLE INDEX

Waste* *Waste Isolation Pilot Plant In Situ Experimental Program for High-Level 000137
Forced-Air Cooling of a Waste Isolation Pilot Plant Mine Drift 000532
Environmental Monitoring at the Waste Isolation Pilot Plant Site, Calendar Year 1976 000263
Revised Concept for the Waste Isolation Pilot Plant 000537
New Mexico, to Problems of Siting the Waste Isolation Pilot Plant* of Fluid Inclusions in Permian Salado Salt, 000072
HWD-Wastes for the Sandia Laboratories Waste Isolation Pilot Project (WIPP), April 1-June 30, 1978* of 000160
High-Level Caustic Waste Solutions Waste Zone Char- Pilot Study* *Removal of Plutonium from 000578
Storage Tanks* *Waste
Minerals* * Pipe Cooling System for Underground, Radioactive Waste 000107
*Radioactive Waste Disposal Report on Sequoyah PIRM Wastes: Permanent Isolation in Rock-Forming 000184
Mine Geologic and Hydrologic Settings, Favorable to Deep Pit Liquid Waste-Shale Column Experiment* 000178
System Below the Zone of Fresh Water, Atlantic Coastal Placement of High-Level Radioactive Wastes*Rock Types, 000060
Aquifers in the Southern Part of the Atlantic Coastal Plain, North Carolina through New Jersey* Aquifer 000366
*Earth Science Technical Plain, South Carolina and Georgia*Potential of Hazardous 000365
Level Waste* * Plan for Mined Geologic Disposal of Radioactive Waste* 000503
Terminal Storage Program, Configuration Management Plan for Solidification of Savannah River Plant High- 000115
Waste Transportation Systems Analysis and Program Plan, Volume 1. Management Summary* *National Waste 000199
Decontamination and Decommissioning, Long Range Plan *Radioactive 000187
Reprocessing Plants* *Long-Term Plan, Idaho National Engineering Laboratory* 000615
*Development of Advanced Waste Management Planning for Management of Aqueous Wastes from Fuel 000523
New Mexico. Special Studies of Waste Isolation Pilot Plans* 000225
with Geologic Associated with the Waste Isolation Pilot Plant (WIPP) Rocks* Plant (WIPP) Site, Southeastern 000073
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site in New Mexico* of Radionuclides 000281
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Geochemistry 000074
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Hydrology* 000056
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Regional 000053
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Seismology* 000055
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Site Geology 000054
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico. Special 000072
Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico* *Geological 000334
Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), December 15, 1978 March 15, 1979* Wastes 000087
Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), October 1-December 15, 1978* Wastes 000088
Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), March 15-June 15, 1979*Transuranic Wastes 000089
Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), October 1-December 15, 1978*Waste Storage 000087
New Mexico Waste Isolation Pilot Plant - A Status Report 000082
the Nuclear Fuel Services, Incorporated, Reprocessing Plant at West Valley, New York* to the Closing of 000164
Potential at the Proposed Waste Isolation Pilot Plant in New Mexico* of Nuclides to Estimate Migration 000380
Underground Waste Disposal Studies, Chemical Processing Plant Area* * 000152
Safety Assessment of the Waste Isolation Pilot Plant Facility* *Preliminary 000088
Looking into the Idaho Chemical Processing Plant Fuel Storage Basin Water* Source of Radionuclides 000184
of Stainless Steels in Idaho Chemical Processing Plant High Level Radioactive Waste Service* Evaluation 000117
Sampling and Analyses of Savannah River Plant High-Level Waste Sludges 000068
Plan for Solidification of Savannah River Plant High-Level Waste 000615
Waste Isolation Pilot Plant In Situ Experimental Program for High-Level Waste 000137
* *Operation of the Monticello Nuclear Generating Plant Liquid and Solid Radioactive Waste Disposal System 000595
Forced-Air Cooling of a Waste Isolation Pilot Plant Mine Drift 000532
the Nuclear Research Center Karlsruhe - Experience from Plant Operation and Development Work* Wastes at 000612
Studies of Concrete as a Host for Savannah River Plant Radioactive Waste 000672
Solid Forms for Savannah River Plant Radioactive Wastes 000682
Environmental Monitoring at the Waste Isolation Pilot Plant Site, Calendar Year 1976 000263
Assay System for the Idaho Chemical Processing Plant Waste Canisters and Spent Fuels Preliminary 000138
Strontium, Plutonium, and Ruthenium from Savannah River Plant Waste Supernate* A Study of Methods for Removing 000688
of Glasses Containing Simulated Savannah River Plant Waste* *Viscosity 000657
Gas Production from Concrete Containing Savannah River Plant Waste* *Radiolytic 000[illegible]
Glass Compositions for Immobilization of Savannah River Plant Waste* *Development of 000[illegible]
as a Matrix for Solidification of Savannah River Plant Waste* *Evaluation of Concrete 000[illegible]
on Borosilicate Glass Containing Savannah River Plant Waste* *Effect of Internal Alpha Radiation 000[illegible]
Microstructures of Glasses Containing Savannah River Plant Waste* of Radionuclide Leachabilities with 000[illegible]
Waste Form for High-Level Idaho Chemical Processing Plant Zirconia Wastes* *Development of a Pelleted 000[illegible]
Geochemistry of Ground Water at the Savannah River Plant 000189
Revised Concept for the Waste Isolation Pilot Plant 000537
Reprocessing Facilities, Idaho Chemical Processing Plant 000583
Waste Package Development at the Rocky Flats Plant* *Radioactive 000136
Waste Expected from a Large Commercial Reprocessing Plant* *Description of Solid 000702
to the Environment from a Nuclear Fuel Reprocessing Plant* *Radioactive Waste Discharges 000622
Radioactive Calcined Waste at Idaho Chemical Processing Plant* *Sampling of Stored High-Level 000158
Zircaloy Solid Wastes from a Fuel Reprocessing Plant* *Management of Highly Active Decladding 000097
Solids Storage Facility at Idaho Chemical Processing Plant* Calcine Retrieved from the Second Calcined 000208
Mexico, to Problems of Siting the Waste Isolation Pilot Plant* of Fluid Inclusions in Permian Salado Salt, New 000072
in Location of a Large Nuclear-Fuel Reprocessing Plant*Importance of Deep Permeable Disposal Formations 000291
of Defense Transuranic Waste at the Savannah River Plant, Aiken, South Carolina* for Long-Term Management 000206
High-Level Radioactive Waste Idaho Chemical Processing Plant, Idaho Falls, Idaho* Term Management of Defense 000346
Retrofitting Trojan Nuclear Plant's Spent Resin Transfer System 000658
the Fate and Behavior of Transuranic Elements in Soils, Plants and Animals* of Microbial Processes to 000210
Radioactivity Released in Effluents from Nuclear Power Plants from 1973 thru 1976* *Summary of 000708
Incorporation of Radioactive Wastes from Nuclear Power Plants into Concrete and Bitumen* 000653
of Neutrition Radioactive Wastes from Fusion Power Plants* *Management 000701
and Movement of Fission Products in Soils and Plants* *Accumulation 000058
for Management of Aqueous Wastes from Fuel Reprocessing Plants* *Long-Term Planning 000523
Formed with Radioactive Wastes at LWR Nuclear Power Plants* of Solidification Agents used and Products 000621
Stress-Strain Data Suitable for Finite-Element Plastic- Plastic Analysis of Shipping Containers* Assessment of 000167
Estimates of Loading on the Sleeve* *Thermoelastic/ Plastic Analysis of Waste-Container Sleeve: 1. Initial 000067
Influence of Salt Strength on Sleeve *Thermoelastic/ Plastic Analysis of Waste-Container Sleeve: 3. 000040
Influence on Hole Closure* *Thermoelastic/ Plastic Analysis of Waste-Container Sleeve: 4. Air Gap 000140
Radioactive Waste* *The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of 000076
Sites: Program Report for the *Cement Technology for Plugging Boreholes in Radioactive Waste Repository 000059
Retention of Plutonium and Americium by Rock 000280
Transuranic Storage Area in Southeastern Idaho* * Plutonium and Americium Contamination near a 000191
Spent Fuel Reprocessing Wastes from the Uranium- Plutonium and Thorium-Uranium Fuel Cycles* Contaminated 000075

PERMUTED TITLE INDEX

ØReview of Criteria for Packaging Urea Char - Pilot Study* Ø Removal of	
Dry Macroporous Resins for the Removal of Uranium and	
ØThe Distribution of	
ØPolonium and	
ØRetention of Organic-Complexed	
Automatic Area Scanning System for the Measurement of	
United States and in Mexico from an ØComparison of	
ØRadiolytic Gas Generation in	
Ground Soil System in Area 11* Ø	
the Consequences of Light Water Reactor Spent Fuel and	
as a Function of pH and Certain Soil ØExtractability of	
Argonne National Laboratory*A Preliminary Assessment of	
Crib Sites* Ø	
Geochemistry* ØAnalysis of	
Technetium, Cesium, Barium, Curium, Europium, Uranium,	
Waste ØA Study of Methods for Removing Strontium,	
ØAre	
and in Deep-Burial Radioactive Waste Disposal* Ø	
Ø	
ØDeformation Mapping of	
of Incorporating Radioactive Wastes in Asphalt or	
ØDispersion in Flow through	
ØFlow in Fractured	
at Low Rayleigh Number from Concentrated Sources in	
ØCalculations of Nuclide Migration in Rock and	
ØMaterial Transport through	
Basal Ottawa Limestone, Chattanooga Shale, Floyd Shale,	
Subsurface Rocks on North Flank of Appalachian Basin Ø	
Structural Syncline in Pennsylvania* Ø	
ØThe Alternatives for Managing	
ØThe Fixation of Radioactive Waste for	
Batch Kd Measurements of Nuclides to Estimate Migration	
ØSubsurface Waste-Disposal	
System Below the ØGeologic Evaluation of Waste-Storage	
Ø	
ØEconomic and Industrial	
and Hydrologic Factors Related to the Waste Storage	
of Nuclear Waste Glass Reactions: Problems and	
Radioactive Waste* ØAntarctica, A	
Clay in Parts of Alabama, Mississippi and Tennessee as	
Radioactive Wastes* ØAntarctica - A	
Ø	
Conditions Expected in the Waste Isolation Pilot Plant Ø	
Conditions Expected in the Waste Isolation Pilot Plant Ø	
Conditions Expected in the Waste Isolation Pilot Plant Ø	
Ø	
Analysis of Hydrologic Conditions Near the Site of a	
ØNational Waste Terminal Storage Program:	
Strain, and Temperature Induced Permeability Changes in	
the Hydraulic Properties of FracturesØObservations of a	
* ØAn Appraisal of Hard Rock for	
* ØAn Appraisal of Hard Rock for	
Anticline, Grand County, Utah* ØSubsurface Geology of a	
ØGeologic Migration	
ØNuclear	
Waste Discharges to the Environment from Nuclear	
Fuel to be Discharged by the United States Nuclear	
of Radioactivity Released in Effluents from Nuclear	
ØIncorporation of Radioactive Wastes from Nuclear	
Management of Nonfritting Radioactive Wastes from Fusion	
Products Formed with Radioactive Wastes at LWR Nuclear	
ØLet's Go Slowly on a Nuclear	
ØThe Management and Storage of Commercial	
ØHandling of Radioactive Waste at Soviet Nuclear	
ØNet Energy from Nuclear	
Nuclear Fuel Cycle for Use in Establishing "As Low As	
Disposal of Liquid ØFoundational Prerequisites and	
ØOperating	
ØOperating	
Reactor Testing Station*	
ØHydrology of Some Deep Mines in	
Isolation in Geologic Formations. Volume 18. Repository	
Isolation in Geologic Formations. Volume 16. Repository	
Isolation in Geologic Formations. Volume 10. Repository	
Isolation in Geologic Formations. Volume 8. Repository	
Isolation in Geologic Formations. Volume 12. Repository	
ØMeasurements of Parameters Essential to	
Waste Glass Reactions: Problems and Potential of	
ØField and Laboratory Studies	
Aspects of Disposal of High-Level Waste in Geologic Ø	
Concentrations in a Stream near Argonne National ØA	
Access Using Underground Mining Techniques for Removal Ø	
for a Nuclear Waste Canister* Ø	
Processing Plant Waste Canisters and Spent Fuels	
Ø	
Fixation of High-Level Radioactive Wastes in Concrete*A	
National Waste Terminal Storage Program* Ø	
of Groundwaters in the Strip Granite: Results and	

Plutonium for Transport by Air*	000052
Plutonium from High-Level Caustic Waste Solutions Using	000578
Plutonium from Waste Streams* ØEvaluation of	000623
Plutonium in a Rock Containment Environment*	000031
Plutonium in an Intertidal Food Chain*	000235
Plutonium in Soil*	000295
Plutonium in Waste* ØAn	000186
Plutonium Concentrations in Deer from the Southeastern	000271
Plutonium Contaminated Waste Materials*	000083
Plutonium Distribution in a Desert Pavement-Desert	000305
Plutonium Shipping Package Loss at Sea* ØEvaluation of	000114
Plutonium 238 and Curium 242 from a Contaminated Soil	000248
Plutonium 239, 240 Concentrations in a Stream near	000201
Plutonium-Americium Soil Penetration at 230-S Building	000170
Plutonium-Release Consequences on the Environmental	000270
Plutonium, and Americium* Initial Studies of Strontium,	000020
Plutonium, and Ruthenium from Savannah River Plant	000088
Plutons the Answer to Nuclear Waste Disposal?*	000310
Pollucite and Its Alteration on Geological Occurrences	000105
Polonium and Plutonium in an Intertidal Food Chain*	000235
Polycrystalline Salt*	000010
Polyethylene* ØThe Feasibility	000596
Porous Media*	000239
Porous Media*	000262
Porous Media* ØSteady Thermal Convection	000037
Porous Media, Penetrated by Water*	000262
Porous Media: A Finite-Element Galerkin Model*	000263
Porters Creek Clay, and Tuscc Clay in Parts of Alabama,	000430
Possibilities for Disposal of Industrial Wastes in	000029
Possibilities for Subsurface Waste Disposal in a	000008
Post-Light-Water Reactor Nuclear Wastes*	000519
Posterity*	000005
Potential at the Proposed Waste Isolation Pilot Plant	000300
Potential in Saline Basin of Kansas*	000394
Potential in Selected Segments of the Roseonic Aquifer	000366
Potential of Denver Basin for Disposal of Liquid Wastes*	000401
Potential of Geologic Traps and Reservoir Strata*	000399
Potential of Roseonic Aquifers in the Southern Part of	000365
Potential of Prediction* ØChemistry	000093
Potential Disposal Site for the World's High-Level	000091
Potential Host Rocks for Underground Emplacement of	000020
Potential International Burial Area for High-Level	000352
Potential Mechanisms of Containment Failure*	000558
Potential Microbial Impact on Transuranic Wastes Under	000007
Potential Microbial Impact on Transuranic Wastes Under	000008
Potential Microbial Impact on Transuranic Wastes Under	000009
Potential Microbial Impact on Transuranic Wastes*	000224
Potential Nuclear Waste Repository, Eddy and Lea	000431
Potential Problems in the Waste Transportation System*	000564
Potential Repository Rocks* ØStress,	000036
Potential Size-Effect in Experimental Determination of	000090
Potential Underground Repositories of Radioactive Wastes	000010
Potential Underground Repositories of Radioactive Wastes	000380
Potential Waste Emplacement Site, Salt Valley	000411
Potentials of Technetium 99 and Neptunium 237*	000411
Power in Sweden*	000322
Power Facilities* ØRadioactive	000280
Power Industry* ØProjections of Spent	000690
Power Plants from 1973 thru 1976* ØSummary	000704
Power Plants into Concrete and Bitumen*	000653
Power Plants* Ø	000701
Power Plants* of Solidification Agents used and	000621
Power Program Until We've Solved Waste Problems*	000527
Power Reactor Wastes*	000681
Power Stations*	000624
Power*	000520
Practicable* Guides - Fabrication of Light-Water	000652
Practice for Using Deep Water-Bearing Horizons for	000293
Practices and Experiences at Hanford*	000179
Practices, Experiences, and Problems at the National	000290
Precambrian Rocks*	000489
Preconceptual Design Studies: Basalt* Radioactive Waste	000549
Preconceptual Design Studies: Battelle Pacific	000550
Preconceptual Design Studies: Granite*Radioactive Waste	000547
Preconceptual Design Studies: Salt* : Radioactive Waste	000545
Preconceptual Design Studies: Shale*: Radioactive Waste	000548
Predicting Nuclide Migration*	000288
Prediction* ØChemistry of Nuclear	000093
Preliminary to Main Experiment*	000012
Preliminary Assessment of the Radiological Protection	000260
Preliminary Assessment of Plutonium 239, 240	000201
Preliminary Concept of Waste Retrieval System-Tank Wall	000630
Preliminary Corrosion Test of a Glass-Ceramic Candidate	000151
Preliminary Description and Operating Procedures Manual*	000138
Preliminary Design of Experiment*	000533
Preliminary Engineering and Economic Analysis of the	000685
Preliminary Geologic Site Selection Factors for the	000488
Preliminary Interpretation* and Isotope Hydrology	000397

PERMUTED TITLE INDEX

Radiostrontium in Barley Soil. Three-Component System* 0 Preliminary Investigations on the Adsorption of 000165
Borehole Measurements at Strip* 0 Preliminary Report on Geophysical and Mechanical 000437
Tested by Solution Mining: 1. Method of Analysis and Preliminary Results for Spherical Cavities* Cavities 000540
On Approach to the Fracture Hydrology at Strip: Preliminary Results* 000398
Scale and Time-Scale Heating Experiments at Strip: Preliminary Results* Wall- 000381
Pilot Plant Facility* 0 Preliminary Safety Assessment of the Waste Isolation 000094
Scale of Admixtures in Preparing Brine Grouts for Radioactive Waste Isolation* 000662
Brines for Disposal of Liquid Preconditions and Practice for Using Deep Water-Bearing 000203
Radioactive Waste* Hot Isostatic Pressing for the Consolidation and Containment of 000529
Permeability Studies at Elevated Temperature and Pressure of Granitic Rocks* Experimental 000069
of Fluids in Rock Salt at Elevated Temperature and Pressure to Repositories for Radioactive Waste* 000080
Corrosion of Metals in a Hot, High Pressure Brine Environment* 000130
in PERMP (Permed Under Elevated Temperature and Pressure) Countries - Experimental Program and Initial 000060
Summary of Previous Pertinent Studies* 000013
Chemical Processing Plant Fuel Storage Basin Water* 0 Primary Source of Radionuclides Leaking Into the Idaho 000144
Storage of Nuclear Migration Paths for Ohio Reactor Products and Applications to the Problem of Geological 000236
Accumulation and Movement of Fission Products in Soils and Plants* 000058
The Behavior of Radioactive Fission Products in Soils* 000272
Characterization of Solidification Agents used and Products Formed with Radioactive Wastes at LRL Nuclear 000621
Vitrnets: Alternatives to Monolithic High Level Waste Products* Clay-Phosphate Ceramics and 000609
Scientific Laboratory* Waste Management Programs and Capabilities within the Los Alamos 000771
Chemical Engineering Division - Fuel Cycle Program Progress Report, January-March 1978* 000536
Waste Management Program Quarterly Report, July-September 1976* 000672
Boreholes in Radioactive Waste Repository Sites: Program Report for the Period October 1, 1977, to 000059
Gulo -- Natural Fission Reactor Program, Program Report for July 1-September 30, 1976* 000065
Analysis for the Nuclear Fuel Cycle-Parts 1 and 2: Program Report for Period April 1 to September 30, 1977 000607
September 30, National Waste Terminal Storage Program Report for Period October 1, 1976, to 000313
National Waste Terminal Storage Program Progress Report June 1977* 000312
Nuclear Waste Management and Transportation Quarterly Progress Report, April Through June 1976* 000333
Chemical Engineering Division Fuel Cycle Program Progress Report, January-March 1978* 000336
Nuclear Waste Management and Transportation Quarterly Progress Report, October Through December 1975* 000226
Waste Radionuclides with Granular Program Approach and Programs* Interaction of 000071
for the Sandia Laboratories Waste Isolation Pilot Project (WIPP), April 1-June 30, 1978* of HAD-Wastes 000160
High Activity Solidified Wastes in Underground Salt 0 Project Salt Vault: A Demonstration of the Disposal of 000316
Theoretical Temperature Fields for the Strip Heater Project, Volume 1* 000378
Annual Report on DOE/ PROJECT 4801954, FY-1975* 000252
United States Nuclear Power Industry* 0 Projections of Spent Fuel to be Discharged by the 000690
Waste Isolation Projects: Fiscal Year 1978* 000487
in Antarctica* Putting Radioactive Waste on Ice - A Proposal for an International Radionuclide Repository 000492
Level Waste Management Description of the Environmental Protection Agency Program for Reprocessing and High 000320
Geologic Preliminary Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste in 000264
the Ocean Floor* Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste on 000261
Considerations of Environmental Protection Criteria for Radioactive Waste* 000526
Fire Protection Device for LWTR Spent Fuel Shipping Casks* 000560
Environmental Protection Requirements for Radioactive Wastes* 000515
and Water Storage* History of Prototype High Level Waste Canister SS-9 While in Air 000101
Flood Basalt* Hypothetical Prototype Sites of Repositories for Radioactive Wastes: 000488
Considerations Relevant to Provisional versus Final Storage* 000559
New York Nuclear Service Center Study, Final Report for Public Comment* Western 000211
* 0 Public Health Implications of Radioactive Waste Releases 000339
Radiological Risks to the Public* 000309
Transuranic Waste Management on the Workers and on the Public* Effects of Current 000209
Separated from High-Level Reprocessing Waste* Final Purification and Concentration of Americium/Curium 000409
for Economic Cooperation and Development Countries Pursue Geological Disposal* Organization 000408
Radioactive Waste* 0 Putting Evaporators to Work: Concentration of Aqueous 000603
International Radionuclide Repository in Antarctica* 0 Putting Radioactive Waste on Ice - A Proposal for an 000492
of the Storage of Radioactive Waste within Silicic Pyroclastic Rocks* An Evaluation 000077
Nuclear Waste Management and Transportation Quarterly Progress Report, April Through June 1976* 000333
* Nuclear Waste Management and Transportation Quarterly Progress Report, October Through December 1975 000226
Waste Management Program Quarterly Report, July-September 1976* 000672
Development, July 1977 Rockwell Hanford Operations Quarterly Report, Process Technology and Process 000335
Savannah River Laboratory Quarterly Report, Waste Management, January-March 1975* 000660
* Savannah River Laboratory Quarterly Report, Waste Management, July-September, 1975 000149
0 Quaternary Studies* 000420
Some Geochemical, Geomechanical, and Hydrogeological Questions* in Argillaceous and Crystalline Rocks: 000218
Statement of Dr. Terry R. Lash* 000511
Degradation of HAD Wastes* 000086
Pilot Experimental Studies of the Degradation of HAD-Wastes for the Sandia Laboratories Waste Isolation 000160
Geologic Persistence* Influence of Radiation on the System Liquid Radioactive Wastes - 000078
River Plant Waste* Effect of Internal Alpha Radiation on Borosilicate Glass Containing Savannah 000099
Salt for Waste Disposal Applications* 0 Radiation Damage Studies on Natural and Synthetic Rock 000081
Salt Utilizing Measurements Made During Electron 0 Radiation Damage Studies on Natural and Synthetic Rock 000082
Waste Receiving Area* 0 Radiation Doses to Rodents Inhabiting a Radioactive 000283
Repositories* Storage and Release of Radiation Energy in Salt is Radioactive Waste 000042
Wastes* Development and Radiation Stability of Glasses for Highly Radioactive 000600
, Part I. Carcinogenic Hazard from Low-Level, Low-Rate Radiation The Argonne Radiological Impact Program (ARIP) 000253
Surfaces* Effects of Radiations from Transuranium Nuclides on Container 000173
Miscellaneous Radioactive and Nonradioactive Wastes* 000695
Autonomous Aspiration Unit for Sampling Radioactive Aerosols from Air* 000182
Solids Storage Facility at Idaho Properties of Radioactive Calcine Retrieved from the Second Calcined 000206
Plant* Sampling of Stored High-Level Radioactive Calcined Waste at Idaho Chemical Processing 000156
Corrosion Monitoring of Storage Bins for Radioactive Calcines* 000116
A Study of the Migration of Radioactive Elements in Soils "In-Place"* 000224
The Behavior of Radioactive Fission Products in Soils* 000272
Fifteen The Leaching of Radioactivity from Highly Radioactive Glass Blocks Buried Below the Water Table: 000685
Leaching of Fully Radioactive High-Level Waste Glass* 000356
The Design of Systems for Handling Radioactive Ion Exchange Resin Beads* 000679
Disposal into a Contamination of Soils: Behavior of Radioactive Ions in Unsaturated Soils after Water 000292

PERMUTED TITLE INDEX

*Solidification and Storage of Hanford's High-Level	Radioactive Liquid Wastes*	000669
*Feasibility of Space Disposal of	Radioactive Nuclear Waste. 1. Executive Summary*	000434
*Feasibility of Space Disposal of	Radioactive Nuclear Waste. 2. Technical Summary*	000433
Ceramic Melter for Vitrification of High-Level *Non-	Radioactive Operation Experience with a Joule Heated	000686
*The Migration of Long-Lived	Radioactive Processing Wastes in Selected Rocks*	000[illegible]0
*Methodology for Estimating Accidental	Radioactive Releases in Nuclear Waste Management*	000278
and Characterization of an Improved High Level	Radioactive Waste (HAW) Borosilicate Glass**Preparation	000605
Using Underground Mining Techniques for Removal of	Radioactive Waste (Sludge, Saltcake and Liquid)* Access	000630
*	Radioactive Waste "Graveyards"*	000485
*Logistics Models for the Transportation of	Radioactive Waste and Spent Fuel*	000566
*Handling of	Radioactive Waste at Soviet Nuclear Power Stations*	000624
*Fixation of	Radioactive Waste by Hydrothermal Reactions with Clays*	000575
*The Fixation of	Radioactive Waste for Posterity*	000405
Physical and Chemical Properties* *Sources of	Radioactive Waste from Light-Water Reactors and	000691
*A Concept for Long-Term Isolation of Solid	Radioactive Waste in Fine-Grained Deposits*	000376
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste in Geologic Formations. Volume 1.	000552
Evaluation of Phosphate and Aerosilicate *Fixation of	Radioactive Waste in Glass. Part 2. The Experimental	000108
*Concerning the Possibility of Disposing of	Radioactive Waste in Ocean Trenches*	000357
International Radionuclide Depository in *Putting	Radioactive Waste on Ice - A Proposal for an	000492
*The Disposal of	Radioactive Waste on Land*	000471
Reactor Concerning the Evolution in the Ground of	Radioactive Waste over a Long Period* Natural Nuclear	000287
*An Evaluation of the Storage of	Radioactive Waste within Silicic Pyroclastic Rocks*	000077
*Interaction Between Processing Method and	Radioactive Waste Burial Site*	000463
*Environmental Assessment Related to Hanford	Radioactive Waste Burial*	000268
*Improved Heat Transfer from	Radioactive Waste Canisters*	000119
Nuclear Power Facilities* *	Radioactive Waste Discharges to the Environment from	000280
Nuclear Fuel Reprocessing Plant* *	Radioactive Waste Discharges to the Environment from a	000633
Glaciological Aspects* *	Radioactive Waste Disposal in Antarctica: Some	000881
Crystalline Rocks: Some *An Appraisal of Underground	Radioactive Waste Disposal in Argillaceous and	000218
*Environmental Aspects of High-Level	Radioactive Waste Disposal in Deep Geologic Formations*	000528
*Long-Term Risk Assessment of	Radioactive Waste Disposal in Geologic Formations*	000256
*	Radioactive Waste Disposal in Geological Formations*	000402
Bibliography with Abstracts* *	Radioactive Waste Disposal in Salt Deposits - A	000836
*A Review of	Radioactive Waste Disposal to the Ground at Hanford*	000191
*United States Seabed High-Level	Radioactive Waste Disposal Feasibility Study*	000308
*Dissolution of Kansas Evaporites - The	Radioactive Waste Disposal Problem*	000859
Waste-Shale Column Experiment* *	Radioactive Waste Disposal Report on Seepage Pit Liquid	000178
Land Burial Water Chemistry at Commercially Low-Level	Radioactive Waste Disposal Sites* of Isotope Migration-	000167
Water Chemistry at Commercially Operated Low-Level	Radioactive Waste Disposal Sites* Migration-Land Burial	000166
Water Chemistry at Commercially Operated Low-Level	Radioactive Waste Disposal Sites* Migration-Land Burial	000168
Monticello Nuclear Generating Plant Liquid and Solid	Radioactive Waste Disposal System* *Operation of the	000595
*Metamict Mineral Alteration: An Implication for	Radioactive Waste Disposal*	000030
*High-Level and Long-Lived	Radioactive Waste Disposal*	000351
*Summary Review of Rock Mechanics Workshop on	Radioactive Waste Disposal*	000372
of an Effective Aluminosilicate Adsorbent for	Radioactive Waste Disposal* *Development	000606
Alteration on Geological Occurrences and in Deep-Burial	Radioactive Waste Disposal* *Pollucite and Its	000105
Elevated Temperature and Pressure) Concretes - *	Radioactive Waste Fixation in FOETAP (Formed Under	000060
*Procedure for Determining Leachabilities of	Radioactive Waste Forms*	000084
*Natural Glasses: Analogues for	Radioactive Waste Forms*	000111
*Leach Rate Characterization of Solid	Radioactive Waste Forms*	000286
Formation from Alpha-Decay and its Significance for	Radioactive Waste Glasses* *Helium	000131
*Technological Facilities at the New Hungarian	Radioactive Waste Handling and Disposal Station*	000593
for Long-Term Management of Defense High-Level	Radioactive Waste Hanford Reservations, Richland,	000387
for Long-Term Management of Defense High-Level	Radioactive Waste Idaho Chemical Processing Plant,	000386
Hanford Groundwater Transport Estimates of Hypothetical	Radioactive Waste Incidents* *	000221
*Operation of Low-Level	Radioactive Waste Incinerator*	000588
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000021
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000022
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000023
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000024
Support for Generic Environmental Impact Statement:	Radioactive Waste Isolation in Geologic Formations.	000075
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000237
Support for Generic Environmental Impact Statement:	Radioactive Waste Isolation in Geologic Formations.	000296
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000388
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000385
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000545
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000546
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000547
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000548
for Generic Environmental Impact Statement (GEIS):	Radioactive Waste Isolation in Geologic Formations.	000549
Support for Generic Environmental Impact Statement:	Radioactive Waste Isolation in Geologic Formations.	000550
Impact Statement on Commercial Waste Management:	Radioactive Waste Isolation in Geologic Formations*	000857
to Dissolution in Salt Domes and How It Affects a	Radioactive Waste Isolation Facility Design* Migration	000263
*Role of Admixtures in Preparing Dense Cements for	Radioactive Waste Isolation*	000662
Development Administration Technological Program for	Radioactive Waste Management and Reprocessing* and	000325
*	Radioactive Waste Management in the United Kingdom*	000613
*	Radioactive Waste Management in Japan*	000509
*The	Radioactive Waste Management Complex and Its Operations*	000208
National Engineering *Safety Review Document for the	Radioactive Waste Management Complex at the Idaho	000176
*	Radioactive Waste Management Development in Europe*	000688
*Guide to	Radioactive Waste Management Literature*	000323
Reactor Testing Station* *High-Level	Radioactive Waste Management Program at the National	000671
Services* *The High-Level	Radioactive Waste Management Program at Nuclear Fuel	000589
Program at Battelle Pacific* *High-Level	Radioactive Waste Management Research and Development	000688
and Application of a Risk Assessment Method for	Radioactive Waste Management. Volume 2. Implementation	000279
*High-Level	Radioactive Waste Management*	000582
Methodology of Environmental Risks Associated with	Radioactive Waste Management* *An Assessment	000259
and Impact Analysis of Separation Methods Applied to	Radioactive Waste Management* *Technology Assessment	000601

PERMUTED TITLE INDEX

Science Underlying	Radioactive Waste Management: Status and Needs*	000663
Barriers*	Radioactive Waste Management: The Need for Multiple	000150
Investigate their Suitability for Possible Storage of	Radioactive Waste Material as of September, 1977.* to	000439
Investigate Their Suitability for Possible Storage of	Radioactive Waste Material* Texas Salt Dome Basin to	000438
Plets Plant*	Radioactive Waste Package Development at the Rocky	000136
	Radioactive Waste Processing*	000476
	Radioactive Waste Processing*	000682
Radiation Doses to Rodents Inhabiting a	Radioactive Waste Receiving Area*	000283
Public Health Implications of	Radioactive Waste Release*	000339
and 2*Mechanical and Thermal Design Considerations for	Radioactive Waste Repositories in Hard Rock. Parts 1	000019
Storage and Release of Radiation Energy in Salt in	Radioactive Waste Repositories*	000042
Geological Criteria for	Radioactive Waste Repositories*	000368
State/Federal Licensing Interactions for	Radioactive Waste Repositories*	000508
of Eastern United States with Regard to Siting of	Radioactive Waste Repositories* Mountain Region	000488
Analysis of the Geological Stability of a Hypothetical	Radioactive Waste Repository in a Bedded Salt Formation*	000562
for Simulating the Deformation Behavior of a High-Level	Radioactive Waste Repository in Bedded Salt* Model	000020
An Evaluation of Erigiag Techniques for High-Level	Radioactive Waste Repository Site Characterization*	000026
the Period Cement Technology for Plugging Boreholes in	Radioactive Waste Repository Sites: Progress Report for	000059
Steels in Idaho Chemical Processing Plant High Level	Radioactive Waste Service* Evaluation of Stainless	000117
Extremely Synthetic Rutile Encapsulation: A	Radioactive Waste Solidification System Resulting in an	000597
Leachabilities of Cesium and Strontium Processing of	Radioactive Waste Solution with Zeolite (2).	000619
Basin with Reference to the Subsurface Disposal of	Radioactive Waste Solutions* Summary of the Appalachian	000379
Crystalline Swedish-American Cooperative Program on	Radioactive Waste Storage in Mined Caverns in	000407
Heat Pipe Cooling System for Underground,	Radioactive Waste Storage Tanks*	000107
Program Plan*	Radioactive Waste Transportation Systems Analysis and	000187
Impact of Waste Effluents in the Correlation of	Radioactive Waste Treatment Costs and the Environmental	000521
Impact of Waste Effluents in the Correlation of	Radioactive Waste Treatment Costs and the Environmental	000652
Environmental Impact of Waste Effluents Correlation of	Radioactive Waste Treatments Costs and the	000501
Atomic Energy Agency (IAEA) Definition of High-Level	Radioactive Waste Unsuitable for Dumping at Sea*	000188
An Improved Receptacle for	Radioactive Waste*	000182
	Radioactive Waste*	060516
Management of Commercial High-Level	Radioactive Waste*	000551
Secure Storage of	Radioactive Waste*	000587
Putting Evaporators to Work: Concentration of Aqueous	Radioactive Waste*	000603
Influence of Surfactants on Bituminization of	Radioactive Waste*	000626
Studies of Concrete as a Host for Savannah River Plant	Radioactive Waste*	000673
Considerations of Environmental Protection Criteria for	Radioactive Waste*	000526
Science Technical Plan for Mined Geologic Disposal of	Radioactive Waste* Earth	000503
A Potential Disposal Site for the World's High-Level	Radioactive Waste* Antarctica,	000491
Pressing for the Consolidation and Containment of	Radioactive Waste* Hot Isostatic	000629
Zinc Borosilicate Glass Containing Simulated High-Level	Radioactive Waste* Volatility of a	000604
for Long-Term Management of Defense High-Level	Radioactive Waste* Summary - Alternatives	000478
Plugging and Shaft Sealing for Geologic Isolation of	Radioactive Waste* The Status of Borehole	000526
Elevated Temperature and Pressure to Repositories for	Radioactive Waste* Chemistry of Fluids in Rock Salt at	000080
Bearing Components from Nuclear Facilities and General	Radioactive Waste* Storage for the Disposal of Non-Fuel	000094
Transportation of	Radioactive Waste-Materials to the Sun*	000488
Geologic Disposal of High-Level	Radioactive Wastes - Earth-Science Perspectives*	000362
Influence of Radiation on the States Liquid	Radioactive Wastes - Geologic Formation*	000078
Karlsruhe - Experience from Plant Bituminization of	Radioactive Wastes at the Nuclear Research Center	000612
of Solidification Agents used and Products Formed with	Radioactive Wastes at LWR Nuclear Power Plants*	000621
Oceanic Research Needed for Safe Disposal of	Radioactive Wastes at Sea*	000449
Management of Nontritium	Radioactive Wastes from Fusion Power Plants*	000701
Concrete and Bitumen* Incorporation of	Radioactive Wastes from Nuclear Power Plants into	000653
Studies on Treating	Radioactive Wastes from Reprocessing Installations*	000625
Recent Results and Developments on the Disposal of	Radioactive Wastes in the Asse Salt Mine*	000177
Vol 4. The Law Concerning the Disposal System Study,	Radioactive Wastes in the Federal Republic of Germany.	000498
in the Federal Republic of Germany: Accumulation of	Radioactive Wastes in the Federal Republic of Germany*	000696
Accumulation of Radioactive Wastes in the Federal	Radioactive Wastes in the Federal Republic of Germany:	000696
for, and Experience With, Underground Storage of Liquid	Radioactive Wastes in the USSR* The Scientific Basis	000484
The Feasibility of Incorporating	Radioactive Wastes in Asphalt or Polyethylene*	000586
Fixation of	Radioactive Wastes in Clay-Flux Mixes*	000682
Long Experiments for Self Fixation of	Radioactive Wastes in Clay-Flux Mixtures*	000617
The Feasibility of Storing	Radioactive Wastes in Columbia River Basalts*	000406
and Economic Analysis of the Fixation of High-Level	Radioactive Wastes in Concrete* Preliminary Engineering	000685
Terminal Storage of	Radioactive Wastes in Geologic Formations*	000425
Disposal of Long-Lived	Radioactive Wastes in Italy*	000603
Disposal of	Radioactive Wastes in Salt Domes*	000038
Evaluation on the Disposal of	Radioactive Wastes into the North Pacific*	000302
Development of Solidification and Burial Methods for	Radioactive Wastes of Fuel Cycle*	000392
the German Democratic Republic* Disposal of	Radioactive Wastes Produced in Nuclear Installations in	000450
Film Evaporators* Volume Reduction of Liquid	Radioactive Wastes Using Mechanically Agitated Thin-	000592
Underground Storage Condition for	Radioactive Wastes*	000005
Durability of Containers for Storing Solidified	Radioactive Wastes*	000085
Retrievable Surface Storage of High-Level	Radioactive Wastes*	000158
Geochemical Behavior of Long-Lived	Radioactive Wastes*	000258
Dispersion Relations in Injection Disposal of	Radioactive Wastes*	000266
Management of	Radioactive Wastes*	000327
	Radioactive Wastes*	000337
	Radioactive Wastes*	000338
Deep Disposal Systems for	Radioactive Wastes*	000355
Geological Criteria for Repositories for High-Level	Radioactive Wastes*	000635
Environmental Protection Requirements for	Radioactive Wastes*	000515
Sea Disposal of Packaged	Radioactive Wastes*	000518
The Brookhaven Experience with Sea Disposal of	Radioactive Wastes*	000598
Microstructures and Leachability of Vitrified	Radioactive Wastes*	000649
Solid Forms for Savannah River Plant	Radioactive Wastes*	000683
Fusion Fuel Cycle Solid	Radioactive Wastes*	000697
and Hydrologic Factors for Locating Repositories for	Radioactive Wastes* Geologic	000827

- A Potential International Burial Area for High-Level and Radiation Stability of Glasses for Highly
of Hard Rock for Potential Underground Repositories of
of Hard Rock for Potential Underground Repositories of
First Barrier for Long-Term Storage of Highly Active
of Underground Storage of Bitumen Blocks Containing
of Germany. Vol 4. The Law Concerning the Disposal of
Radiometric Control at the Stage of Transporting Solid
of Gulf Coast Salt Domes for the Storage or Disposal of
Safety Assessment of Geologic Disposal of Commercial
Settings, Favorable to Deep Placement of High-Level
Deep Water-Bearing Horizons for Disposal of Liquid
and Western Oklahoma for Underground Storage of
#Hypothetical Prototype Sites of Repositories for
#Investigation of Cooling Coil Corrosion in
Buried Below the Water Table: Fifteen #The Leaching of
#Studies on Fixation of
#Fixation of
#Fixation of
Level Waste# #Calculations of
Plants from 1973 thru 1976# #Summary of
#Fixation of
Casks for Spent Fuel and High-Level #Calculations of
Carcinogenic Hazard from Low-Level, Low- #The Argonne
Sea# #
Level Waste on the Ocean Floor# #Assessment of the
Level Waste in Geologic #Preliminary Assessment of the
#
Underground Storage of Bitumen# #
Hydrogenous Waste# #Gas Generation from
Waste Materials# #
Nuclear Wastes# #
Savannah River Plant Waste# #
#
Radioactive #Experience in the Organization of Sanitary
Waste on Ice - A Proposal for an International
Produced by the Cannikin #Hydrologic Processes and
Volume 1: Processes Influencing Radionuclide Mobility #
Glasses Containing Savannah River Plant #Correlation of
Overview# #
Formation (Application to #Mathematical Modelling of
Site in Southern New Mexico# #
#The Effect of Retardation Factors on
#Geochemistry and
Soil and Rock Media. Volume 1: Processes Influencing
#A Transport Model for Long Term Release of Low-Level
#Interactions of Simulated Waste
#Influences of Marine Sediment on the Accumulation of
#Selection of Significant Elements and
#Diffusion of
Transportation Fires# #Release of
#Oceanic Distribution of
Application of Mass Transport Theory to the Leaching of
#Sorption and Migration of
Considerations Underlying the Migration of
Part 2: #Information Pertinent to the Migration of
#Variables Affecting Sorption and Transport of
#Critical Pathways of
Teclation Pilot Plant (WIPP) Site in #Interaction of
Progress# #Interaction of Waste
Processing Plant Fuel Storage Basin #Primary Source of
Theoretical Estimation of Strontium 90 Migration# #
Concentration Limits for Shallow Land Burial of
Interactions of Geologic Media with Waste
#Preliminary Investigations of the Adsorption of
#Movement of
#Incorporation of High-Level
#Costs for
#Transportation of Spent Fuel by Barge and By
#
Theory and Numerical Implementation (Discrete-Parcel-
of High Sodium-Alumina Wastes: Composition
#Leach
(ABIP), Part I. Carcinogenic Hazard from Low-Level, Low-
Americium, and Curium in #Determination of Distribution
Waste# #Neutron and Gamma-
Media# #Steady Thermal Convection at Low
#Mineral
#Hydrothermal Glass
#Studies on the Hydrolysis
#Fixation of Radioactive Waste by Hydrothermal
#Waste-Rock Interactions and Bedrock
#Chemistry of Nuclear Waste Glass
Catalysed Waste-Rock Interactions# #Hydrothermal
#Evaluation of the Activity Levels in Fusion
Drawn from Studies on the Okio Natural Nuclear
#Contaminated
Low As Practicable" Guides - Fabrication of Light-Water

Radioactive Wastes# #Antarctica 000352
Radioactive Wastes# #Development 000608
Radioactive Wastes# #An Appraisal 000018
Radioactive Wastes# #An Appraisal 000380
Radioactive Wastes# #Glass as the 000128
Radioactive Wastes# #Feasibility Study 000581
Radioactive Wastes# Wastes in the Federal Republic 000495
Radioactive Wastes# in the Organization of Sanitary 000567
Radioactive Wastes# #An Investigation of the Utility 000428
Radioactive Wastes# #The United States Program for the 000319
Radioactive Wastes# Types, Also Geologic and Hydraulic 000068
Radioactive Wastes#Prerequisites and Practice for Using 000203
Radioactive Wastes#Salt Deposits in the Texas Panhandle 000417
Radioactive Wastes: Flood Basalt# 000444
Radioactive-Waste Storage Tanks# 000139
Radioactivity from Highly Radioactive Glass Blocks 000645
Radioactivity in Fired Clays# 000678
Radioactivity in Stable, Solid Media# 000646
Radioactivity in Stable, Solid Media# 000680
Radioactivity Release Due to Leaching of Vitrified High 000591
Radioactivity Released in Effluents from Nuclear Power 000704
Radioisotopes on Hanford Soil Types# 000183
Radiological Consequences from Sabotage of Shipping 000586
Radiological Impact Program (ABIP), Part I. 000253
Radiological Problems of Liquid Waste Disposal into the 000201
Radiological Protection Aspects of Disposal of High- 000261
Radiological Protection Aspects of Disposal of High- 000264
Radiological Risks to the Public# 000309
Radiolysis and Temperature Effects in Case of 000202
Radiolytic Attack of Transuranic-Contaminated 000091
Radiolytic Gas Generation in Plutonium Contaminated 000043
Radiolytic Gas Production during Long-Term Storage of 000577
Radiolytic Gas Production from Concrete Containing 000099
Radiolytic Gas Production from Tritiated Waste Forms# 000100
Radiometric Control at the Stage of Transporting Solid 000567
Radionuclide Depository in Antarctica# Radioactive 000492
Radionuclide Distribution in a Cavity and Chimney 000377
Radionuclide Interaction with Soil and Rock Media. 000217
Radionuclide Leachabilities with Microstructures of 000620
Radionuclide Migration from Low Level Waste: A Generic 000302
Radionuclide Migration in a Homogeneous Clayey 000291
Radionuclide Migration Studies Associated with the WIPP 000240
Radionuclide Migration# 000257
Radionuclide Migration# 000264
Radionuclide Mobility and Retention; Element Chemistry 000217
Radionuclide Solutions into a Stream Ecosystem# 000311
Radionuclides and Rocks# 000039
Radionuclides by Green Alga (Ulva pertusa)# 000285
Radionuclides for Waste Management Assessments# 000703
Radionuclides from Geologic Formations# 000234
Radionuclides from High-Level Waste During 000110
Radionuclides from Nuclear Weapons Testing# 000230
Radionuclides from Waste Solids# # 000259
Radionuclides in Geologic Media# 000244
Radionuclides in Geomedia: Okio and Other Examples# 000231
Radionuclides in Ground Water at the Nevada Test Site - 000163
Radionuclides in Hanford Subsoils# 000354
Radionuclides to Man from Agro-Ecosystems# 000255
Radionuclides with Geomedia Associated with the Waste 000241
Radionuclides with Geomedia: Program Approach and 000071
Radionuclides Leaking into the Idaho Chemical 000188
Radionuclides Migration in Aerated Zones. 3. 000288
Radionuclides# #Suggested 000277
Radionuclides# #Microstructural 000038
Radiostrontium in Saclay Soil. Three-Component System# 000165
Radiostrontium in Soils# 000366
Radwaste in SYNROC# 000145
Radwaste Treatment# 000502
Rail: A Comparison of Risk# 000565
Raman Spectra of Cesium Aluminosilicate Waste Forms# 000113
Random-Walk Version)# Mass Transport Model: 000215
Ranges and Properties# #Vitrification 000104
Rate Characterization of Solid Radioactive Waste Forms# 000286
Rate Radiation#The Argonne Radiological Impact Program 000253
Ratios and Diffusion Coefficients of Neptunium, 000301
Ray Sources in Light Water Reactor High-Level Nuclear 000590
Rayleigh Number from Concentrated Sources in Porous 000037
Reaction Work at Hanford# 000572
Reactions in Salt Brine# 000155
Reactions of Metal Ions# 000006
Reactions with Clays# 000575
Reactions# 000089
Reactions: Problems and Potential of Prediction# 000093
Reactivity of Simulated Nuclear Waste Forms and Water- 000058
Reactor Blankets. Waste Problem# 000699
Reactor Concerning the Evolution in the Ground of 000247
Reactor Facilities# 000692
Reactor Fuel from Enriched Uranium Dioxide# "As 000652

PERMUTED TITLE INDEX

Cycle - Reprocessing of High-Temperature Gas-Cooled
@Neutron and Gamma-Ray Sources in Light Water
@The Alternatives for Managing Post-Light-Water
of Non-Fuel Bearing Hardware Resulting from Light Water
Geological Storage of Nuclear @Migration Paths for Ohio
30, 1979@ @Ohio — Natural Fission
at Sea@ @Evaluation of the Consequences of Light Water
for the Transportation of Liquid Metal Fast Breeder
@Tennessee Valley Authority Away-From-
Fuel Storage Requirements - The Need for Away-From-
Practices, Experiences, and Problems at the National
Radioactive Waste Management Program at the National
@Hydrology of Waste Disposal National
@Leaching Characteristics of Actinides from Simulated
Characterization of Solidified Simulated
@Light-Water
@The Management and Storage of Commercial Power
@Sources of Radioactive Waste from Light-Water
@Carbon 14 Production in Nuclear
of Some Geologic and Ocean Disposal Concepts Regarding
@Shipments of Nuclear Fuel and Waste...Are They
National Waste Terminal Storage Program@ @As Low as is
Doses to Rodents Inhabiting a Radioactive Waste
@An Improved
@Mechanically Agitated Thin-Film Evaporators@ @Volume
from the Nuclear Fuel Cycle: January-March 1978@@Volume
from the Nuclear Fuel Cycle: July-September 1978@@Volume
the Nuclear Fuel Cycle: October-December 1977@ @Volume
Non-Fuel Bearing Hardware @Experience in Volume
and Serpentinite Bodies from the Appalachian Mountain
Investigation Underground Limestone Mines in the Ohio
@
@
Pilot Plant (WIPP) Site, Southeastern New Mexico.
Management@ @Nuclear
Risk Assessment Review Group Report to the U.S. Nuclear
States Geological Survey Studies of Physical-Chemical
1967@ @A Review of Studies on Soil-Waste
Stream Ecosystem@ @A Transport Model for Long Term
Waste Repositories@ @Storage and
Transportation Fires@ @
@Analysis of Plutonium-
@Calculations of Radioactivity
Repository@ @Computer Enhanced "
1973 thru 1976@ @Summary of Radioactivity
@Methodology for Estimating Accidental Radioactive
@Public Health Implications of Radioactive Waste
Waste Management@ @
@Considerations
Savannah River Plant Waste @A Study of Methods for
@Mine
@Geological Criteria for
in Rock Salt at Elevated Temperature and Pressure to
@Geologic and Hydrologic Factors for Locating
@Hypothetical Prototype Sites of
and Thermal Design Considerations for Radioactive Waste
@An Appraisal of Hard Rock for Potential Underground
@An Appraisal of Hard Rock for Potential Underground
Wastes in Geological Formations@ @Site Selection for
@Geological Criteria for Radioactive Waste
@Long-Term Safety Assessment of Geologic Isolation
Federal Licensing Interactions for Radioactive Waste
of Radiation Energy in Salt in Radioactive Waste
Studies of Physical-Chemical Relationships in Salt
States with Regard to Siting of Radioactive Waste
Masses from Stored Nuclear Waste in Natural Rock
2. Implementation for Terminal Storage in Reference
Stability of a Hypothetical Radioactive Waste
@A Waste
Deformation Behavior of a High-Level Radioactive Waste
@Information Base for Waste
@Deep Exploratory Test Holes for Waste
Interactions of High Active Waste with Crystalline Rock
for the Suitability of Anhydrite as a Nuclear Waste
Waste Isolation in Geologic Formations. Volume 16.
Waste Isolation in Geologic Formations. Volume 16.
Waste Isolation in Geologic Formations. Volume 10.
Waste Isolation in Geologic Formations. Volume 9.
Waste Isolation in Geologic Formations. Volume 12.
Temperature Induced Permeability Changes in Potential
of Kriging Techniques for High-Level Radioactive Waste
@Selection and Description of Generic
Technology for Plugging Boreholes in Radioactive Waste
and Accountability Procedures for a Waste Isolation
"Release Scenario" Analysis for a Nuclear Waste
the MOL Data Structure in Consequence Analysis for WIPP
- Retrieval, Processing, and Shipment to the Federal
of the Rock Mechanics Aspects of the Proposed Salt Mine

Reactor Fuel Containing Uranium 233 and Thorium@ Fuel 000581
Reactor High-Level Nuclear Waste? 000590
Reactor Nuclear Wastes@ 000519
Reactor Operations@ Transportation, and Disposal 000651
Reactor Products and Applications to the Problem of 000236
Reactor Program, Progress Report for July 1-September 000065
Reactor Spent Fuel and Plutonium Shipping Package Loss 000118
Reactor Spent Fuel Rods@@Evaluation of Organic Coolants 000096
Reactor Spent Fuel Storage Studies@ 000320
Reactor Storage@ @Spent 000705
Reactor Testing Station@ @Operating 000200
Reactor Testing Station@ @High-Level 000671
Reactor Testing Station, Idaho, An Interim Report@ 000410
Reactor Waste Glass@ 000153
Reactor Wasteform@ @Microstructural 000115
Reactor Wastes to be Isolated in Geologic Formations@ 000146
Reactor Wastes@ 000681
Reactors and Physical and Chemical Properties@ 000691
Reactors@ 000694
Realistic Modeling that Allows Objective Risk 000517
Really Safe?@ 000152
Reasonably Achievable (ALARA) Studies Relative to the 000493
Receiving Area@ @Radiation 000283
Receptacle for Radioactive Waste@ 000182
Reduction of Liquid Radioactive Wastes Using 000592
Reduction System for Solid and Liquid Transuranic Waste 000570
Reduction System for Solid and Liquid Transuranic Waste 000571
Reduction System for Solid and Liquid TRU Waste from 000638
Reduction, Packaging, Transportation, and Disposal of 000651
Region of Eastern United States with Regard to Siting 000484
Region@ @Report of 000370
@ Regional Geology of the Northeast Texas Salt Dome Basin@ 000412
@ Regional Geology@ 000485
Regional Geology@ Report, Waste Isolation 000453
Regulatory Commission Program for High Level Waste 000508
Regulatory Commission@ @ 000512
Relationships in Salt Repositories@ @United 000079
Relationships on the Hanford Reservation from 1944 to 000197
Release of Low-Level Radionuclide Solutions into a 000311
Release of Radiation Energy in Salt in Radioactive 000042
Release of Radionuclides from High-Level Waste During 000110
Release Consequences on the Environmental Geochemistry@ 000270
Release Due to Leaching of Vitrified High Level Waste@ 000591
Release Scenario" Analysis for a Nuclear Waste 000560
Released in Effluents from Nuclear Power Plants from 000704
Releases in Nuclear Waste Management@ 000270
Releases@ 000339
Relevance of Natural Retention Experience to Nuclear 000588
Relevant to Provisional versus Final Storage@ 000559
Removing Strontium, Plutonium, and Ruthenium from 000688
Renovation and Preparation of Experimental Area@ 000395
Repositories for High-Level Radioactive Wastes@ 000435
Repositories for Radioactive Waste@ Chemistry of Fluids 000080
Repositories for Radioactive Wastes@ 000427
Repositories for Radioactive Wastes: Flood Basalt@ 000488
Repositories in Hard Rock. Parts 1 and 2@ @Mechanical 000019
Repositories of Radioactive Wastes@ 000019
Repositories of Radioactive Wastes@ 000380
Repositories of Solid High-Level and Alpha-Bearing 000413
Repositories@ 000368
Repositories@ 000369
Repositories@ @State/ 000504
Repositories@ @Storage and Release 000042
Repositories@ @United States Geological Survey 000079
Repositories@ Mountain Region of Eastern United 000484
Repositories@ on Accumulation of Actinide Critical 000015
Repository and Other Applications@ Management. Volume 000279
Repository in a Bedded Salt Formation@of the Geological 000562
Repository in a Geologic Formation@ 000529
Repository in Bedded Salt@ Model for Simulating the 000020
Repository Design, Volume 6. Excavation Technology@ 000542
Repository Investigations@ 000426
Repository Media@ @Long Term Geochemical 000157
Repository Media@ and Brine and Their Implications 000016
Repository Preconceptual Design Studies: Basalt@ 000589
Repository Preconceptual Design Studies: Battelle@ 000550
Repository Preconceptual Design Studies: Granite@ 000587
Repository Preconceptual Design Studies: Salt@ 000585
Repository Preconceptual Design Studies: Shale@ 000588
Repository Rocks@ @Stress, Strain, and 000036
Repository Site Characterization@ @An Evaluation 000026
Repository Sites@ 000458
Repository Sites: Progress Report for the Period 000059
Repository@ @Material Control 000120
Repository@ @Computer Enhanced 000560
Repository@ @The Utilization of 000223
Repository@ @Alternatives Involving 000381
Repository@ @Analysis and Evaluation 000174

and Accountability Considerations for a Waste Isolation
Irradiation on Migration of Brine Inclusions in a Salt
Conditions Near the Site of a Potential Nuclear Waste
Storage Facilities* $Design Aspects of the Alpha
$Ion Exchange Properties of a
of the Environmental Protection Agency Program for
of Technetium 99 and Selenium 79 in Nuclear Fuel
of Carbon 14, Krypton 85 and Iodine 129 in Nuclear Fuel
Impact of Waste Effluents in the Nuclear Fuel Cycle -
and Fixation for Waste Management, Production, and
* $
$Studies on Treating Radioactive Wastes from
the Closing of the Nuclear Fuel Services, Incorporated,
of Solid Waste Expected from a Large Commercial
Waste Discharges to the Environment from a Nuclear Fuel
Active Decladding Zircaloy Solid Wastes from a Fuel
Disposal Formations in Location of a Large Nuclear-Fuel
Planning for Management of Aqueous Wastes from Fuel
Nuclear Fuel and Final Storage of Vitrified High-Level
of Americium/Curium Separated from High-Level
of High-Level and Actinide-Contaminated Spent Fuel
Program for Radioactive Waste Management and
$System Study, Radioactive Wastes in the Federal
Development Program on Waste Disposal in the Federal
Accumulation of Radioactive Wastes in the Federal
in the Federal $Radioactive Wastes in the Federal
in Nuclear Installations in the German Democratic
Program for Radioactive $Description of the Energy
$High-Level Radioactive Waste Management
Laboratory* $High-Level Waste Management
the Federal Republic of Germany* $The
$Ion Exchange
$Oceanographic
$Bituminization of Radioactive Wastes at the Nuclear
at Sea* $Oceanic
of Studies on Soil-Waste Relationships on the Hanford
Geology and Ground-Water Characteristics of the Hanford
of Cable Mountain Basalt Cores Hanford Nuclear
of Defense High-Level Radioactive Waste Hanford
$Subsurface Disposal in Geologic Basins - A Study of
and Industrial Potential of Geologic Basins and
Design of Systems for Handling Radioactive Ion Exchange
$Retrofitting Trojan Nuclear Plant's Spent
Waste Streams* $Evaluation of New Macroporous
Nuclear Fuel* $Corrosion
and How It Affects a Radioactive Waste Isolation $The
$The Effect of
$Sorption and
$
$
$Relevance of Natural
1: Processes Influencing Radionuclide Mobility and
$Disposition (D and D) of
Acceptable Temperatures of Wastes and Containers During
Wastes* $
National Engineering Laboratory* $
$System Design for
$High-Level Waste Canister Emplacement and
$Initial Drum
$Early Waste
Mining Techniques for $Preliminary Concept of Waste
National Engineering Laboratory* $Alternatives Involving
Repository* $Alternatives Involving
Facility at Idaho $Properties of Radioactive Calcine
Transfer System* $
Site of a Potential Nuclear Waste Repository, Eddy and $
Ridge, Tennessee* $A
$Nuclear Waste Management--
Transport by Air* $
Hanford* $A
Disposal* $Summary
Hanford Reservation from 1944 to 1967* $A
Complex at the Idaho National Engineering $Safety
Commission* $Risk Assessment
$National Waste Terminal Storage
Subsurface in Russell, Lincoln, Ellsworth, Barton, and
High-Level Radioactive Waste Hanford Reservations,
$Waste Disposal by Shale Fracturing at Oak
$Waste Disposal by Shale Fracturing at Oak
Management Research and Development Program at Oak
$Sorption and Retention by Clay Minerals; Oak
$A Review of the Near Surface Heater Experiment at Oak
Geologic Formations* $Long-Term
Regarding Realistic Modeling that Allows Objective
Management. Volume 2. $Development and Application of a
Regulatory Commission* $
of Spent Fuel by Barge and By Rail: A Comparison of
$Radiological

Repository* $An Overview of Material Control 000496
Repository* Temperature Gradients, Stress, and 000041
Repository, Eddy and Lea Counties, New Mexico* of Hydrologic 000431
Repository: 2. Conceptual Layouts of Underground 000541
Representative Oil Sand* 000029
Reprocessing and High Level Waste Management* 000328
Reprocessing and Waste Solidification Facility Off-Gas* 000598
Reprocessing and Waste Solidification Off-Gas* 000245
Reprocessing of High-Temperature Gas-Cooled Reactor 000501
Reprocessing Division of the Department of Energy (July- 000440
Reprocessing Facilities, Idaho Chemical Processing Plant 000583
Reprocessing Installations* 000625
Reprocessing Plant at West Valley, New York* Related to 000169
Reprocessing Plant* $Description 000702
Reprocessing Plant* $Radioactive 000633
Reprocessing Plant* $Management of Highly 000097
Reprocessing Plant* $Importance of Deep Permeable 000391
Reprocessing Plants* $Long-Term 000523
Reprocessing Waste. 4. Safety Analysis* of Spent 000123
Reprocessing Waste* Final Purification and Concentration 000409
Reprocessing Wastes from the Uranium-Plutonium and 000035
Reprocessing* Development Administration Technological 000325
Republic of Germany. Vol 4. The Law Concerning the 000395
Republic of Germany* $The Research and 000161
Republic of Germany* in the Federal Republic of Germany: 000696
Republic of Germany: Accumulation of Radioactive Wastes 000696
Republic* $Disposal of Radioactive Wastes Produced 000450
Research and Development Administration Technological 000725
Research and Development Program at Battelle Pacific* 000644
Research and Development Program at Oak Ridge National 000579
Research and Development Program on Waste Disposal in 000161
Research by the United States Geological Survey* 000064
Research on Waste Disposal Off the Coast of Tokai-Mura* 000415
Research Center Karlsruhe - Experience from Plant 000612
Research Needed for Safe Disposal of Radioactive Wastes 000469
Reservation from 1944 to 1967* $A Review 000197
Reservation of the U.S. Atomic Energy Commission, 000189
Reservation* $Thermal Properties 000053
Reservations, Richland, Washington* Long-Term Management 000367
Reservoir Strata* 000400
Reservoir Strata* $Economic 000399
Resin Beads* $The 000670
Resin Transfer System* 000658
Resins for the Removal of Uranium and Plutonium from 000623
Resistance of Canisters for Final Disposal of Spent 000132
Response of Salt Migration to Dissolution in Salt Domes 000263
Retardation Factors on Radionuclide Migration* 000257
Retention by Clay Minerals: Oak Ridge Studies* 000675
Retention of Organic-Complexed Plutonium in Soil* 000295
Retention of Plutonium and Americium by Rock* 000289
Retention Experience to Nuclear Waste Management* 000588
Retention; Element Chemistry and Geochemistry; 000217
Retired Contaminated Facilities at Hanford* 000175
Retrievable Geologic Storage* $Maximum 000121
Retrievable Surface Storage of High-Level Radioactive 000158
Retrieval of Buried Transuranic Waste at the Idaho 000184
Retrieval of Solidified High-Level Wastes at Hanford* 000212
Retrieval Concepts Study* 000561
Retrieval Final Report* 000185
Retrieval Interim Report* 000186
Retrieval System-Tank Hall Access Using Underground 000630
Retrieval, Processing, and Disposal on the Idaho 000342
Retrieval, Processing, and Shipment to the Federal 000341
Retrieved from the Second Calcined Solids Storage 000208
Retrofitting Trojan Nuclear Plant's Spent Resin 000658
Review and Analysis of Hydrologic Conditions Near the 000431
Review of the Near Surface Heater Experiment at Oak 000421
Review of Commercial Nuclear Waste Management* 000610
Review of Criteria for Packaging Plutonium for 000092
Review of Radioactive Waste Disposal to the Ground at 000191
Review of Rock Mechanics Workshop on Radioactive Waste 000373
Review of Studies on Soil-Waste Relationships on the 000197
Review Document for the Radioactive Waste Management 000176
Review Group Report to the U.S. Nuclear Regulatory 000512
Review Meeting on Salt Dissolution* 000348
Rice Counties, Central Kansas* Part 1: Salt Beds in the 000679
Richland, Washington* Long-Term Management of Defense 000347
Ridge National Laboratory* 000213
Ridge National Laboratory* 000214
Ridge National Laboratory* $High-Level Waste 000579
Ridge Studies* 000675
Ridge, Tennessee* 000821
Risk Assessment of Radioactive Waste Disposal in 000256
Risk Assessment to be Made* and Ocean Disposal Concepts 000517
Risk Assessment Method for Radioactive Waste 000279
Risk Assessment Review Group Report to the U.S. Nuclear 000512
Risk* $Transportation 000565
Risks to the Public* 000309

PERMUTED TITLE INDEX

*An Assessment Methodology of Environmental	Risks Associated with Radioactive Waste Management*	000250
Feasibility of Storing Radioactive Wastes in Columbia	River Basalts* *The	000406
July-September, 1975* *Savannah	River Laboratory Quarterly Report, Waste Management,	000199
January-March 1975* *Savannah	River Laboratory Quarterly Report, Waste Management,	000668
*Sampling and Analyses of Savannah	River Plant High-Level Waste Sludges*	000468
*Plan for Solidification of Savannah	River Plant High-Level Waste*	000615
*Studies of Concrete as a Host for Savannah	River Plant Radioactive Waste*	000673
*Solid Forms for Savannah	River Plant Radioactive Wastes*	000683
Strontium, Plutonium, and Ruthenium from Savannah	River Plant Waste Supernate* of Methods for Removing	000688
*Viscosity of Glasses Containing Simulated Savannah	River Plant Waste*	000657
of Concrete as a Matrix for Solidification of Savannah	River Plant Waste* *Evaluation	000467
Gas Production from Concrete Containing Savannah	River Plant Waste* *Radiolytic	000098
of Glass Compositions for Immobilization of Savannah	River Plant Waste* *Development	000658
Radiation on Borosilicate Glass Containing Savannah	River Plant Waste* *Effect of Internal Alpha	000099
with Microstructures of Glasses Containing Savannah	River Plant Waste* of Radionuclide Leachabilities	000620
*Geochemistry of Ground Water at the Savannah	River Plant*	000180
Management of Defense Transuranic Waste at the Savannah	River Plant, Aiken, South Carolina* for Long-Term	000206
Nevada to *Geologic Investigation of the Virgin	River Valley Salt Deposits, Clark County, Southeastern	000429
*Calculations of Nuclide Migration in	Rock and Porous Media, Penetrated by Water*	000262
Radioactive Wastes* *An Appraisal of Hard	Rock for Potential Underground Repositories of	000019
Radioactive Wastes* *An Appraisal of Hard	Rock for Potential Underground Repositories of	000180
*The Distribution of Plutonium in a	Rock Containment Environment*	000037
*Waste-	Rock Interactions and Bedrock Reactions*	000059
*Waste-	Rock Interactions*	000082
Simulated Nuclear Waste Forms and Water-Catalyzed Waste-	Rock Interactions* *Hydrothermal Reactivity of	000058
*Deformation of a Jointed	Rock Mass*	000466
Repository* *Analysis and Evaluation of the	Rock Mechanics Aspects of the Proposed Salt Mine	000178
*Summary Review of	Rock Mechanics Workshop on Radioactive Waste Disposal*	000373
Radionuclide *Radionuclide Interaction with Soil and	Rock Media. Volume 1: Processes Influencing	000217
Isolation in Geologic Formations. Volume 7. Baseline	Rock Properties - Basalt* (GEIS): Radioactive Waste	000023
Isolation in Geologic Formations. Volume 6. Baseline	Rock Properties - Shale* (GEIS): Radioactive Waste	000022
Waste Isolation in Geologic Formations. Baseline	Rock Properties--Granite* Statement (GEIS): Radioactive	000021
Isolation in Geologic Formations. Volume 4. Baseline	Rock Properties: Salt* (GEIS): Radioactive Waste	000388
Critical Masses from Stored Nuclear Waste in Natural	Rock Repositories* on Accumulation of Actinide	000015
Interactions of High Active Waste with Crystalline	Rock Repository Media* *Long Term Geochemical	000157
*Application of Physical Chemistry of Fluids in	Rock Salt at Elevated Temperature and Pressure to	000080
*Radiation Damage Studies on Natural and Synthetic	Rock Salt for Waste Disposal Applications*	000081
*Thermal Conductivity of	Rock Salt from Louisiana Salt Domes*	000061
*Mechanical Behavior of New Mexico	Rock Salt in Triaxial Compression up to 200 C*	000076
*Radiation Damage Studies on Natural and Synthetic	Rock Salt Utilizing Measurements Made During Electron	000082
*	Rock Stresses Associated with Burial of Nuclear Waste*	000085
of the Rocks in the Bureau of Mines Standard	Rock Suite* *Thermal Conductivity	000063
Facility Construction Feasibility and Costs by	Rock Type* Waste Isolation in Geologic Formations.	000548
Favorable to Deep Placement of High-Level Radioactive	Rock Types, Also Geologic and Hydraulic Settings,	000068
*Analysis of Some Tracer Runs in Granite	Rock Using a Fissure Model*	000287
*Fracture Detection in Crystalline	Rock Using Ultrasonic Shear Waves*	000881
for Radioactive Waste Repositories in Hard	Rock. Parts 1 and 2* and Thermal Design Considerations	000019
*Retention of Plutonium and Americium by	Rock*	000249
*Interactions between Nuclear Waste and Surrounding	Rock*	000282
of a Glass-Based Nuclear Waste Simulant and Natural	Rock* *Interaction	000040
Waste by In-Situ Incorporation in Deep Molten Silicate	Rock* *Disposal of Nuclear	000632
*PIPP Wastes: Permanent Isolation in	Rock-Forming Minerals*	000189
*Kinetic Model for Silicate	Rock-Ground Water Interaction*	000088
Geologic Media* *The	Rock-Salt Approach to Nuclear Waste Disposal in	000017
Waste Storage in Mined Caverns in Crystalline	Rock, Program Summary* Program on Radioactive	000887
*In Situ Heating Experiments in Hard	Rock: Their Objectives and Design*	000382
of Alabama, Mississippi and Tennessee as Potential Host	Rocks for Underground Emplacement of Waste* in Parts	000630
*Thermal Conductivity of the	Rocks in the Bureau of Mines Standard Rock Suite*	000063
*Compressible Fluid Flow Through	Rocks of Variable Permeability*	000066
for Disposal of Industrial Wastes in Subsurface	Rocks on North Flank of Appalachian Basin in New York*	000820
*Interactions of Simulated Waste Radionuclides and	Rocks*	000030
*Pine-Grained Sheet Silicate	Rocks*	000883
*Hydrology of Some Deep Mines in Precambrian	Rocks*	000889
of Long-Lived Radioactive Processing Wastes in Selected	Rocks* *The Migration	000250
on Nuclide Migration in Fractured Crystalline	Rocks* *In-Situ Experiments	000276
Storage of Radioactive Waste within Silicic Pyroclastic	Rocks* *An Evaluation of the	000077
and Neptunium 237 in Contact with Unweathered Igneous	Rocks* *Chemistry of Technetium 99	000228
Induced Permeability Changes in Potential Repository	Rocks* *Stress, Strain, and Temperature	000036
at Elevated Temperature and Pressure of Granitic	Rocks* *Experimental Permeability Studies	000069
Special Studies of Waste Isolation Pilot Plant (WIPP)	Rocks*Pilot Plant (WIPP) Site, Southeastern New Mexico.	000072
Waste Disposal in Argillaceous and Crystalline	Rocks: Some Geochemical, Geomechanical, and	000218
Experimental Evaluation of Waste Transport in Selected	Rocks: 1977 Annual Report of Lawrence Berkeley	000219
Technology and Process Development, July 1977 through *	Rockwell Hanford Operations Quarterly Report, Process	000335
*Radioactive Waste Package Development at the	Rocky Flats Plant*	000136
*Radiation Doses to	Rodents Inhabiting a Radioactive Waste Receiving Area*	000282
of Liquid Metal Fast Breeder Reactor Spent Fuel	Rods* of Organic Coolants for the Transportation	000096
Mining Methods Used in the Underground Tunnels and Test	Rooms at Strips*	000350
*Analysis of Some Tracer	Runs in Granite Rock Using a Fissure Model*	000287
Central Kansas, Part 1: Salt Beds in the Subsurface in	Russell, Lincoln, Ellsworth, Barton, and Rice Counties,	000470
Study of Methods for Removing Strontium, Plutonium, and	Ruthenium from Savannah River Plant Waste Supernate* *A	000688
*Confinement of Volatilized	Ruthenium Oxides*	000427
Solidification System Resulting in an *Synthetic	Rutile Microencapsulation: A Radioactive Waste	000597
of the Idaho National Engineering Laboratory (Including	RWMC, Site 98, and Lenhi Range)* *Present Environment	000207
Measured Values for the State of Stress in the Earth'	s Crust* *An Analysis of the	000818
Solved Waste Problem* *Let'	s Go Slowly on a Nuclear Power Program Until We've	000527
*Solidification and Storage of Hanford'	s High-Level Radioactive Liquid Wastes*	000669
*Antarctica, A Potential Disposal Site for the World'	s High-Level Radioactive Waste*	000891
*Underground Disposal of Canada'	s Nuclear Waste*	000870

PERMUTED TITLE INDEX

©The Management of Canada'	s Nuclear Wastes®	000317
©Retrofitting Trojan Nuclear Plant®	s Spent Basin Transfer System®	000658
©Plutonium-Americium Soil Penetration at 238-	S Building Crib Sites®	000170
Characteristics of the Hanford Reservation of the U.	S. Atomic Energy Commission, Washington® Ground-Water	000189
1977® ©Fluidized Bed Incineration System for U.	S. Department of Energy Defense Waste, July-December	000349
©Adsorption Studies by the U.	S. Geological Survey®	000576
©Risk Assessment Review Group Report to the U.	S. Nuclear Regulatory Commission®	000512
Wastes® ©U.	S. Program for the Immobilization of High-Level Nuclear	000586
Dry Underground Storage Space in the Midcontinent, U.	S.A.® Search for Carbonates Suitable for Developing	000371
Level ©Calculations of Radiological Consequences from	Sabotage of Shipping Casks for Spent Fuel and High-	000506
Investigations of the Adsorption of Radiostrontium in	Saline Soil. Three-Component System® ©Preliminary	000165
©Oceanic Research Needed for	Safe Disposal of Radioactive Wastes at Sea®	000049
©Shipments of Nuclear Fuel and Waste...Are They Really	Safe?®	000152
©Nuclear	Safety and the Environment®	000332
Scenario Analysis Methods for Use in Assessing the	Safety of the Geologic Isolation of Nuclear Waste®	000502
Storage of Vitrified High-Level Reprocessing Waste. 4.	Safety Analysis® of Spent Nuclear Fuel and Final	000123
©The Waterproof Geometry of Salt Domes, An Important	Safety Aspect for their Exploitation for Ultimate	000407
Facility® ©Preliminary	Safety Assessment of the Waste Isolation Pilot Plant	000498
Radioactive Wastes® ©The United States Program for the	Safety Assessment of Geologic Disposal of Commercial	000319
©Long-Term	Safety Assessment of Geologic Isolation Repositories®	000369
Laboratory Contract No. 85901AK. Waste Isolation	Safety Assessment Program - Collection and Generation	000219
Term Characterization Task ©Overview of Waste Isolation	Safety Assessment Program and Description of Source	000315
Management Complex at the Idaho National Engineering ®	Safety Review Document for the Radioactive Waste	000176
©Application of Studies of Fluid Inclusions in Permian	Salado Salt, New Mexico, to Problems of Siting the	000072
©Subsurface Waste-Disposal Potential in	Saline Basin of Kansas®	000398
Calculating Effects of Liquid Waste Disposal in Deep	Saline Aquifers. Part 1 -- Development, Part 2 --	000267
©Application of Physical Chemistry of Fluids in Rock	Salt at Elevated Temperature and Pressure to	000080
©Radiation Damage Studies on Natural and Synthetic Rock	Salt for Waste Disposal Applications®	000081
©Thermal Conductivity of Rock	Salt from Louisiana Salt Domes®	000061
©Storage and Release of Radiation Energy in	Salt in Radioactive Waste Repositories®	000082
©Mechanical Behavior of New Mexico Rock	Salt in Triaxial Compression up to 200 C®	000086
©Estimates of Relative Areas for the Disposal in Bedded	Salt of LWR Wastes from Alternative Fuel Cycles®	000050
in Oil and Gas Test Holes in Central Kansas, Part 1:	Salt Beds in the Subsurface in Russell, Lincoln,	000479
©Hydrothermal Glass Reactions in	Salt Brine®	000155
Analysis and Preliminary ©Structural Analysis of	Salt Cavities Formed by Solution Mining: 1. Method of	000540
Detroit-Mine Experiment® ©Growth and Shape of	Salt Cavity--Computer Model and Calibration from	000462
©Radioactive Waste Disposal in	Salt Deposits - A Bibliography with Abstracts®	000436
Oklahoma for Underground Storage ©Evaluation of Permi	Salt Deposits in the Texas Panhandle and Western	000417
©Geologic Investigation of the Virgin River Valle,	Salt Deposits, Clark County, Southeastern Nevada to	000339
Kansas, Part 1: Salt Beds in the Subsurface in ®	Salt Dissolution in Oil and Gas Test Holes in Central	000479
®	Salt Dissolution in Oil and Gas Test Holes in Kansas®	000478
©National Waste Terminal Storage Review Meeting on	Salt Dissolution®	000388
Study of the Interior Salt Domes of Northeast Texas	Salt Dome Basin to Investigate Their Suitability for	000432
©Regional Geology of the Northeast Texas	Salt Dome Basin®	000412
Genesis and Growth of Salt Domes in the Northeast Texas	Salt Dome Basin®	000432
©The Response of Salt Migration to Dissolution in	Salt Domes and How It Affects a Radioactive Waste	000263
Wastes® ©An Investigation of the Utility of Gulf Coast	Salt Domes for the Storage or Disposal of Radioactive	000428
©Genesis and Growth of	Salt Domes in the Northeast Texas Salt Dome Basin®	000432
Investigate Their ©Geologic Study of the Interior	Salt Domes of Northeast Texas Salt Dome Basin to	000438
©Disposal of Radioactive Wastes in	Salt Domes®	000038
©Thermal Conductivity of Rock Salt from Louisiana	Salt Domes®	000061
©Geohydrologic Studies of Gulf Coast Interior	Salt Domes®	000460
Exploitation for Ultimate ©The Waterproof Geometry of	Salt Domes, An Important Safety Aspect for their	000407
a Hypothetical Radioactive Waste Repository in a Bedded	Salt Formation® Analysis of the Geological Stability of	000562
Storage of High-Level Waste Solids and Liquids in	Salt Formations® ©Ultimate	000359
Storage of High-Level Waste Solids and Liquids in	Salt Formations® ©Ultimate	000360
Affects a Radioactive Waste Isolation ©The Response of	Salt Migration to Dissolution in Salt Domes and How It	000263
of the Rock Mechanics Aspects of the Proposed	Salt Mine Repository® ©Analysis and Evaluation	000176
on the Disposal of Radioactive Wastes in the Mine	Salt Mine® ©Recent Results and Developments	000177
of High Activity Solidified Wastes in Underground	Salt Mines® Salt Vault: A Demonstration of the Disposal	000316
®	Salt Motion Following Nuclear Waste Disposal®	000386
Survey Studies of Physical-Chemical Relationships in	Salt Repositories® ©United States Geological	000070
and Irradiation on Migration of Brine Inclusions in a	Salt Repository® Temperature Gradients, Stress,	000041
Analysis of Waste-Container Sleeve: 3. Influence of	Salt Strength on Sleeve Loading® ©Thermoelastic/Plastic	000440
into Fresh Water Aquifer. 1. ©Investigation of	Salt Transport in Vertical Boreholes and Brine Invasion	000273
into Fresh Water Aquifer. 2. ©Investigation of	Salt Transport in Vertical Boreholes and Brine Invasion	000274
into Fresh Water Aquifer. 3. ©Investigation of	Salt Transport in Vertical Boreholes and Brine Invasion	000275
©Radiation Damage Studies on Natural and Synthetic Rock	Salt Utilizing Measurements Made During Electron	000082
Geology of a Potential Waste Emplacement Site,	Salt Valley Anticline, Grand County, Utah® ©Subsurface	000411
Activity Solidified Wastes in Underground Salt ©Project	Salt Vault: A Demonstration of the Disposal of High	000316
©Deformation Mapping of Polycrystalline	Salt®	000010
Volume 8. Repository Preconceptual Design Studies:	Salt® Waste Isolation in Geologic Formations.	000585
Battelle Pacific Northwest Laboratory Waste Forms in	Salt® 16. Repository Preconceptual Design Studies:	000550
of a High-Level Radioactive Waste Repository in Bedded	Salt® Model for Simulating the Deformation Behavior	000020
Formations. Volume 4. Baseline Rock Properties:	Salt® (GEIS): Radioactive Waste Isolation in Geologic	000384
©Geology of Certain Domes of the Northeast Texas	Salt-Dome Basin®	000353
Isolation in Geologic Formations. Stratigraphics of	Salt, Granite, Shale, and Basalt® : Radioactive Waste	000385
of Studies of Fluid Inclusions in Permian Salado	Salt, New Mexico, to Problems of Siting the Waste	000072
Techniques for Removal of Radioactive Waste (Sludge,	Saltcake and Liquid)® Access Using Underground Mining	000630
level Waste Sludges® ®	Sampling and Analyses of Savannah River Plant High-	000460
Waste at Idaho Chemical Processing Plant® ®	Sampling of Stored High-Level Radioactive Calcined	000158
©Autonomous Aspiration Unit for	Sampling Radioactive Aerosols from Air®	000182
©Sedimentary History and Economic Geology of	San Juan Basin, New Mexico and Colorado®	000443
©Ion Exchange Properties of a Representative Oil	Sand®	000029
Studies of the Degradation of HAD-Wastes for the	Sandia Laboratories Waste Isolation Pilot Project (WIPP)	000160
Waste Solidification Method® ©The	Sandia Solidification Process - A Broad Range Aqueous	000636
Transporting Solid ©Experience in the Organization of	Sanitary Radiometric Control at the Stage of	000567
Management, July-September, 1975® ®	Savannah River Laboratory Quarterly Report, Waste	000199

PERMUTED TITLE INDEX

Management, January-March 1975* $ Savannah River Laboratory Quarterly Report, Waste 000660
@Sampling and Analyses of Savannah River Plant High-Level Waste Sludges* 000648
@Glass for Solidification of Savannah River Plant High-Level Waste* 000615
@Studies of Concrete as a Host for Savannah River Plant Radioactive Waste* 000673
@Solid Forms for Savannah River Plant Radioactive Waste* 000683
for Removing Strontium, Plutonium, and Ruthenium from Savannah River Plant Waste Separator's Study of Methods 000688
@Radiolytic Gas Production from Concrete Containing Savannah River Plant Waste* 000098
@Viscosity of Glasses Containing Simulated Savannah River Plant Waste* 000657
Development of Glass Compositions for Immobilization of Savannah River Plant Waste* $ 000658
of Concrete as a Matrix for Solidification of Savannah River Plant Waste* @Evaluation 000067
Alpha Radiation on Borosilicate Glass Containing Savannah River Plant Waste* @Effect of Internal 000099
with Microstructures of Glasses Containing Savannah River Plant Waste* Radionuclide Leachabilities 000620
@Geochemistry of Ground Water at the Savannah River Plant* 000180
Term Management of Defense Transuranic Waste at the Savannah River Plant, Aiken, South Carolina* for Long- 000206
@Nuclear Waste: Increasing Scale and Sociopolitical Impacts* 000510
Preliminary Results* @Full- Scale and Time-Scale Heating Experiments at Stripa: 000381
* @Full-Scale and Time- Scale Heating Experiments at Stripa: Preliminary Results 000381
Waste* @An Automatic Drum Scanning System for the Measurement of Plutonium in 000106
Safety of the Geologic Isolation of Nuclear @Program Scenario Analysis Methods for Use in Assessing the 000542
@Hanford Groundwater Scenario Studies* 000222
@Computer Enhanced "Plumes Scenario" Analysis for a Nuclear Waste Repository* 000560
Disposal of High-Level Radioactive Wastes - Earth- Science Perspectives* @Geologic 000362
Radioactive Waste* @Earth Science Technical Plan for Final Geologic Disposal of 000503
and Rocks* $ Science Underlying Radioactive Waste Management: Status 000663
$ Scientific Basis for Nuclear Waste Management* 000330
@Low-Level Solidified Waste Characterization: The Scientific Basis for Physical and Chemical Testing* 000003
Storage of Liquid Radioactive Wastes in the USSR* @The Scientific Basis for, and Experience With, Underground 000064
Progress and Capabilities within the Los Alamos Scientific Laboratory* @Waste Management 000171
Capabilities for Alpha Bearing Wastes at the Los Alamos Scientific Laboratory* @Waste Management 000419
$ Sea Disposal of Packaged Radioactive Wastes* 000518
@The Brookhaven Experience with Sea Disposal of Radioactive Wastes* 000598
Radiological Problems of Liquid Waste Disposal into the Seas* $ 000281
Needed for Safe Disposal of Radioactive Wastes at Seas* @Oceanic Research 000449
High-Level Radioactive Waste Unsuitable for Dumping at Seas* Atomic Energy Agency (IAEA) Definition of 000108
Spent Fuel and Plutonium Shipping Package Loss at Seas* of the Consequences of Light Water Reactor 000118
Feasibility Study* @United States Seabed High-Level Radioactive Waste Disposal 000308
@Nuclear Wastes in the Deep Seabed* 000356
@The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive Waste* 000536
Material, Volume 1 and 2* @Feasibility of Sealing Boreholes with Compacted Natural Earthen 000052
Materials, Volume 3* @Feasibility of Sealing Boreholes with Compacted Natural Earthen 000066
Underground Storage Space in the Midcontinent, @Cursory Search for Carbonates Suitable for Developing Dry 000371
$ Secure Storage of Radioactive Waste* 000587
1977 to September 1978 Mathematical Simulation of Sediment and Contaminant Transport in Surface Waters* 000289
Alga (Ulva pertusa)* @Influences of Marine Sediment on the Accumulation of Radionuclides by Green 000285
Basin, New Mexico and Colorado* $ Sedimentary History and Economic Geology of San Juan 000483
at Hanford* @Characterization of Actinide-Bearing Sediments Underlying Liquid Waste Disposal Facilities 000290
@Radioactive Waste Disposal Report on Seepage Pit Liquid Waste-Shale Column Experiment* 000178
Evaluation of Waste-Storage Potential in Selected Segments of the Mesozoic Aquifer System Below the Zone 000366
@Operational Techniques to Minimize, Segregate, and Account for Alpha-Bearing Wastes* 000618
Pilot Plant (WIPP) Site, Southeastern New Mexico. Seismology* Characterization Report, Waste Isolation 000455
of Long-Lived Radioactive Processing Wastes in Selected Rocks* @The Migration 000250
and Experimental Evaluation of Waste Transport in Selected Rocks: 1977 Annual Report of Lawrence Berkeley 000219
the @Geologic Evaluation of Waste-Storage Potential in Selected Segments of the Mesozoic Aquifer System Below 000366
$ Selection and Description of Generic Repository Sites* 000458
Alpha-Bearing Wastes in Geological Formations* @Site Selection for Repositories of Solid High-Level and 000413
Transuranic Waste Incinerator* $ Selection of a Criticality Monitoring Technique for a 000650
Waste Management Assessment* $ Selection of Significant Elements and Radionuclides for 000703
Storage Program* @Preliminary Geologic Site Selection Factors For the National Waste Terminal 000488
@Gold and Selective Storage Solves Nuclear Waste Problem* 000159
and Development for the Monitoring of Technetium 99 and Selenium 79 in Nuclear Fuel Reprocessing and Waste 000598
Sisterna* @Heat Experiments for Self Fixation of Radioactive Wastes in Clay-Flux 000617
Deformation @Interim Report on Development of a Semi-Empirical Numerical Model for Simulating the 000020
Purification and Concentration of Americium/Curium Separated from High-Level Reprocessing Waste* @Final 000409
Management@Technology Assessment and Impact Analysis of Separation Methods Applied to Radioactive Waste 000601
@Waste Management Programs Quarterly Report, July- September 1976* 000672
Technology and Process Development, July 1977 through September 1977* Operations Quarterly Report, Process 000335
and Immobilization for Nuclear Facilities (April- September 1977)* @Waste Incineration 000616
Contaminant Transport in @Annual Report October 1977 to September 1978 Mathematical Simulation of Sediment and 000289
Transuranic Waste from the Nuclear Fuel Cycle: July- September 1978* Reduction System for Solid and Liquid 000571
Program Progress Report for Period October 1, 1976, to September 30, 1977* @National Waste Terminal Storage 000313
Parts 1 and 2: Progress Report for Period April 1 to September 30, 1977* Analysis for the Nuclear Fuel Cycle- 000647
Progress Report for the Period October 1, 1977, to September 30, 1978* Radioactive Waste Repository Sites: 000059
Fission Reactor Program, Progress Report for July 1- September 30, 1979* @Ohio -- Natural 000065
Laboratory Quarterly Report, Waste Management, July- September, 1975* @Savannah River 000199
Possible Storage of Radioactive Waste Material as of September, 1977.* to Investigate their Suitability for 000439
Cyclone Drum Incinerator and Treatment System: July- September, 1978* @TRU Waste 000622
@Supplementary Report: Geologic Feasibility of Talc and Serpentinite Bodies from the Appalachian Mountain 000088
@Western New York Nuclear Service Center Study, Final Report for Public Comment* 000211
Samples from Test Trench at Western New York Nuclear Service Center, West Valley, New York* Analysis of Soil 000173
Chemical Processing Plant High Level Radioactive Waste Service* Evaluation of Stainless Steels in Idaho 000117
@Description of Allied-General Nuclear Services On-Site Solid Waste Storage Concepts* 000205
Radioactive Waste Management Program at Nuclear Fuel Services* @The High-Level 000589
@Issues Related to the Closing of the Nuclear Fuel Services, Incorporated, Reprocessing Plant at West 000169
@Measurement and Control of Cement Set Times in Waste Solidification* 000674
Radioactive @Rock Types, Also Geologic and Hydraulic Settings, Favorable to Deep Placement of High-Level 000068
Waste* @The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive 000536
Waste Disposal Report on Seepage Pit Liquid Waste- Shale Column Experiment* @Radioactive 000178
@Sorption and Fixation of Large Cations by Shale Formations* 000018

PERMUTED TITLE INDEX

	*Waste Disposal by	Shale Fracturing at Oak Ridge National Laboratory* 000213
	*Waste Disposal by	Shale Fracturing at Oak Ridge National Laboratory* 000214
Johnson Method*	*Flowability of Bulk Solids Used in	Shale Fracturing Grouts as Determined by the Jenike and 000641
	*Thermophysical Properties of Conasauga	Shale* 000076
	*Thermal Calculations in	Shale* 000562
Volume 12. Repository Preconceptual Design Studies:		Shale* Waste Isolation in Geologic Formations. 000508
Formations. Volume 6. Baseline Rock Properties -		Shale* (GRIS): Radioactive Waste Isolation in Geologic 000022
Geologic Formations. Stratigraphics of Salt, Granite,		Shale, and Basalt*GRIS): Radioactive Waste Isolation in 000285
in Parts of	*Basal Ottawa Limestone, Chattanooga	Shale, Floyd Shale, Porters Creek Clay, and Yazoo Clay 000830
	*Basal Ottawa Limestone, Chattanooga Shale, Floyd	Shale, Porters Creek Clay, and Yazoo Clay in Parts of 000830
	Methods for Determining the Thermal History of	Shales* *Geothermometry 000087
	*Thermal Effects in	Shales: Measurements and Modeling* 000057
	*Suggested Concentration Limits for	Shallow Land Burial of Radionuclides* 000277
Fracture Detection in Crystalline Rock Using Ultrasonic		Shear Waves* 000091
	*Fine-Grained	Sheet Silicate Rocks* 000482
	*Alternatives Involving Retrieval, Processing, and	Shipment to the Federal Repository* 000341
Safe?*		Shipments of Nuclear Fuel and Waste...Are They Really 000152
	*Nuclear	Shipping and Waste Disposal Cost Estimates* 000507
	*Development of High-Level Waste	Shipping Cask Models* 000112
	of Radiological Consequences from Sabotage of	Shipping Casks for Spent Fuel and High-Level Waste* 000506
	*Fire Protection Device for LMFBR Spent Fuel	Shipping Casks* 000560
Suitable for Finite-Element Plastic-Plastic Analysis of		Shipping Containers*An Assessment of Stress-Strain Data 000183
	of Light Water Reactor Spent Fuel and Plutonium	Shipping Package Loss at Sea* of the Consequences 000114
Management Assessment*	*Selection of	Significant Elements and Radionuclides for Waste 000703
	*Bonding of Metal Oxides in Sodium	Silicate Glass* 000126
Nuclear Waste by In-Situ Incorporation in Deep Molten		Silicate Rock* *Disposal of 000632
	*Kinetic Model for	Silicate Rock-Ground Water Interaction* 000088
	*Fine-Grained Sheet	Silicate Rocks* 000483
Evaluation of the Storage of Radioactive Waste within		Silicic Pyroclastic Rocks* *An 000077
	*Interaction of a Glass-Based Nuclear Waste	Simulant and Natural Rock* 000040
	*Stabilization of	Simulated Commercial Waste Calcine* 000660
*Volatility of a Zinc Borosilicate Glass Containing		Simulated High-Level Radioactive Waste* 000608
Optical	*Characterization of Leached Surface Layers on	Simulated High-Level Waste Glasses by Sputter-Induced 000118
Rock Interactions*	*Hydrothermal Reactivity of	Simulated Nuclear Waste Forms and Water-Catalysed Waste- 000054
	*Leaching Characteristics of Actinides from	Simulated Reactor Waste Glass* 000153
	*Microstructural Characterization of Solidified	Simulated Reactor Wasteforms* 000115
	*Viscosity of Glasses Containing	Simulated Savannah River Plant Waste* 000657
	*Interactions of	Simulated Waste Radionuclides and Rocks* 000039
on Development of a Semi-Empirical Numerical Model for		Simulating the Deformation Behavior of a High-Level 000020
Brine Invasion into Fresh Water Aquifer. 3. Numerical		Simulation of Brine Invasion of Fresh Water Aquifers* 000275
	Report October 1977 to September 1978 Mathematical	Simulation of Sediment and Contaminant Transport in 000289
	of Radionuclides in Ground Water at the Nevada Test	Site - Part 2: Annotated Bibliography* to the Migration 000163
	*Antarctica, A Potential Disposal	Site for the World's High-Level Radioactive Waste* 000891
Associated with the Waste Isolation Pilot Plant (WIPP)		Site in New Mexico* of Radionuclides with Geomedia 000241
Radionuclide Migration Studies Associated with the WIPP		Site in Southern New Mexico* *An 000280
	*Review and Analysis of Hydrologic Conditions Near the	Site of a Potential Nuclear Waste Repository, Eddy and 000831
Techniques for High-Level Radioactive Waste Repository		Site Characterization* *An Evaluation of Kriging 000026
	Pilot Plant (WIPP) Site, Southeastern New Mexico.	Site Geology* Characterization Report, Waste Isolation 000058
Alpha-Bearing Wastes in Geological Formations*		Site Selection for Repositories of Solid High-Level and 000813
Storage Program*	*Preliminary Geologic	Site Selection Factors For the National Waste Terminal 000488
	*Description of Allied-General Nuclear Services On-	Site Solid Waste Storage Concepts* 000205
Idaho National Engineering Laboratory (Including NRWC,		Site 10, and Lemhi Range)* *Present Environment of the 000207
	*Heater Experiments in the Climax Stock, Nevada Test	Site* 000486
Between Processing Method and Radioactive Waste Burial		Site* *Interaction 000863
	Monitoring at the Waste Isolation Pilot Plant	Site, Calendar Year 1976* *Environmental 000363
	*Petrology of Tuff Units from the J-13 Drill	Site, Jackass Flats, Nevada* 000410
	*Subsurface Geology of a Potential Waste Emplacement	Site, Salt Valley Anticline, Grand County, Utah* 000411
	Report, Waste Isolation Pilot Plant (WIPP)	Site, Southeastern New Mexico. Geochemistry* 000078
	Report, Waste Isolation Pilot Plant (WIPP)	Site, Southeastern New Mexico. Hydrology* 000456
	Report, Waste Isolation Pilot Plant (WIPP)	Site, Southeastern New Mexico. Regional Geology* 000453
	Report, Waste Isolation Pilot Plant (WIPP)	Site, Southeastern New Mexico. Seismology* 000455
	Report, Waste Isolation Pilot Plant (WIPP)	Site, Southeastern New Mexico. Site Geology* 000454
	Report, Waste Isolation Pilot Plant (WIPP)	Site, Southeastern New Mexico. Special Studies of Waste 000073
	Report, Waste Isolation Pilot Plant (WIPP)	Site, Southeastern New Mexico* Characterization 000338
	of Eleana Argillite from Area 17, Nevada Test	Site, Using the Transient Method* the Permeability 000088
Basalt*	*Hypothetical Prototype	Sites of Repositories for Radioactive Wastes: Flood 000484
	*Selection and Description of Generic Repository	Sites* 000058
	Americium Soil Penetration at 234-5 Building Crib	Sites* *Plutonium- 000170
	Operated Low-Level Radioactive Waste Disposal	Sites* Land Burial Water Chemistry at Commercially 000166
	Operated Low-Level Radioactive Waste Disposal	Sites* Land Burial Water Chemistry at Commercially 000168
	at Commercially Low-Level Radioactive Waste Disposal	Sites* of Isotope Migration-Land Burial Water Chemistry 000167
for Plugging Boreholes in Radioactive Waste Repository		Sites: Program Report for the Period October 1, 1977, 000059
Mountain Region of Eastern United States with Regard to		Siting of Radioactive Waste Repositories* Appalachian 000488
	in Permian Salado Salt, New Mexico, to Problems of	Siting the Waste Isolation Pilot Plant*Fluid Inclusions 000072
	*Waste Isolation Pilot Plant In	Site Experimental Program for High-Level Waste* 000137
Crystalline Rocks*	*In-	Site Experiments on Nuclide Migration in Fractured 000276
and Design*	*In	Site Heating Experiments in Hard Rock: Their Objectives 000382
	*Disposal of Nuclear Waste by In-	Site Incorporation in Deep Molten Silicate Rock* 000632
Evaluation of Nuclear Waste Disposal by Underground In		Site Melting* *Economic and Environmental 000498
	of Active and Inactive Mines for Possible Use as In	Site Test Facilities* *Survey 000379
	*WIPP In-	Site Test Needs Workshop, August 29-30, 1978* 000383
	Container Sleeve: 3. Influence of Salt Strength on	Sleeve Loading*Thermoelastic/Plastic Analysis of Waste- 000440
	Sleeve: 1. Initial Estimates of Loading on the	Sleeve* Plastic Analysis of Waste-Container 000067
	*Thermoelastic/Plastic Analysis of Waste-Container	Sleeve: 1. Initial Estimates of Loading on the Sleeve* 000067
	*Thermoelastic/Plastic Analysis of Waste-Container	Sleeve: 3. Influence of Salt Strength on Sleeve Loading* 000440
	*Thermoelastic/Plastic Analysis of Waste-Container	Sleeve: 4. Air Gap Influence on Hole Closure* 000140
	Mining Techniques for Removal of Radioactive Waste (	Sludge, Saltcake and Liquid)* Access Using Underground 000630
	and Analyses of Savannah River Plant High-level Waste	Sludges* *Sampling 000468

PERMUTED TITLE INDEX

@Nuclear Waste: Increasing Scale and
@Bonding of Metal Oxides in
Properties* @Vitrification of High
@Montmorillonite Exchange Equilibria with Strontium-
Radionuclide Mobility @Radionuclide Interaction with
ox Plutonium 238 and Curium 242 from a Contaminated
a Contaminated Soil as a Function of pH and Certain
@Plutonium-Americium
Nuclear Services Center, West @Geotechnical Analysis of
Distribution in a Desert Pavement-Desert Bound
@Fixation of Radioisotopes on Hanford
of the Adsorption of Radiostrontium in Saclay
@Retention of Organic-Complexed Plutonium in
@Waste Management of Actinide Contaminated
Coefficients of Neptunium, Americium, and Curium in
from 1948 to 1967* @A Review of Studies on
@A Study of the Migration of Radioactive Elements in
of Soils: Behavior of Radioactive Ions in Unsaturated
@Accumulation and Movement of Fission Products in
@Migration of Strontium 90 and Curium 244 in
@The Movement of Liquid through
@The Behavior of Radioactive Fission Products in
@Movement of Radiostrontium in
Sorption of Microquantities of Strontium and Cesium in
to the Fate and Behavior of Transuranic Elements in
Soils after Water Disposal into a *Contamination of
Fuel Cycle: January-March @Volume Reduction System for
Fuel Cycle: July-September @Volume Reduction System for
October-December 1977* @Volume Reduction System for
@
Formations* @Site Selection for Repositories of
@Fixation of Radioactivity in Stable,
@Fixation of Radioactivity in Stable,
@A Concept for Long-Term Isolation of
of the Monticello Nuclear Generating Plant Liquid and
@Leach Rate Characterization of
@Fusion Fuel Cycle
Radiometric Control at the Stage of Transporting
Reprocessing Plant* @Description of
@Long-Term Elevated Temperature Leaching of
@Description of Allied-General Nuclear Services On-Site
@Management of Highly Active Decladding Zircaloy
Wastes of Fuel Cycle* @Development of
Radioactive Liquid Wastes* @
and Cesium in Intermediate-Level Liquid Wastes by
@Techniques for High-Level Waste
Drum Drier* @PIPS - A Process for the
@Plan for
@Evaluation of Concrete as a Matrix for
Radioactive Wastes @Physicochemical Characterization of
and Selenium 79 in Nuclear Fuel Reprocessing and Waste
Solidification Process - A Broad Range Aqueous Waste
and Iodine 129 in Nuclear Fuel Reprocessing and Waste
Solidification Method* @The Sandia
Rutile Microencapsulation: A Radioactive Waste
@Current High-Level Waste
@Measurement and Control of Cement Set Times in Waste
a Titanate Based Ceramic for High Level Nuclear Waste
@System Design for Retrieval of
@Durability of Containers for Storing
@Microstructural Characterization of
for Physical and Chemical Testing* @Low-Level
Vault: A Demonstration of the Disposal of High Activity
@Ultimate Storage of High-Level Waste
@Ultimate Storage of High-Level Waste
Radioactive Calcine Retrieved from the Second Calcined
the Jenike and Johanson Method* @Flowability of Bulk
Theory to the Leaching of Radionuclides from Waste
Level Waste* @A Possible
Strontium from @Processing of Radioactive Waste
Results @Structural Analysis of Salt Cavities Formed by
Model for Long Term Release of Low-Level Radionuclide
@Ion Exchange Decontamination of Alkaline
- A Process for the Solidification of Fission Product
@Removal of Plutonium from High-Level Caustic Waste
to the Subsurface Disposal of Radioactive Waste
@Let's Go Slowly on a Nuclear Power Program Until We've
on the Treatment of Medium Level Wastes and Spent
@Gold and Selective Storage
@Evaluation of Cesium 137
Formations* @
Media* @
Studies* @
Subsoils* @Variables Affecting
Soils* @The
241 on Two Subsoils from Differing Weathering @
@Americium 241, Neptunium 237, and Technetium 99
@Illite: A Model Based on the

Sociopolitical Topics* 000510
Sodium Silicate Glass* 000126
Sodium-Aluminum Wastes: Composition Ranges and 000108
Sodium-Cesium* 000027
Soil and Rock Media. Volume 1: Processes Influencing 000217
Soil as a Function of pH and Certain Soil Components: 000208
Soil Components: $BaO-SeO_2$ System* and Curium 242 from 000208
Soil Penetration at 234-5 Building Crib Sites* 000170
Soil Samples from Test Trench at Western New York 000173
Soil System in Area 11* @Plutonium 000305
Soil Types* 000183
Soil. Three-Component System*Preliminary Investigations 000165
Soil* 000295
Soil* 000408
Soil-Aquatic Environments* Ratios and Diffusion 000301
Soil-Waste Relationships on the Hanford Reservation 000197
Soils "In-Place"* 000229
Soils after Water Disposal into a Limited Surface, Part 000292
Soils and Plants* 000058
Soils of Different Textures* 000227
Soils* 000265
Soils* 000272
Soils* 000306
Soils* @The 000047
Soils, Plants and Animals* of Microbial Processes 000310
Soils: Behavior of Radioactive Ions in Unsaturated 000292
Solid and Liquid Transuranic Waste from the Nuclear 000570
Solid and Liquid Transuranic Waste from the Nuclear 000571
Solid and Liquid TRU Waste from the Nuclear Fuel Cycle: 000638
Solid Forms for Savannah River Plant Radioactive Wastes* 000683
Solid High-Level and Alpha-Bearing Wastes in Geological 000413
Solid Media* 000686
Solid Media* 000680
Solid Radioactive Waste in Pine-Trained Deposits* 000376
Solid Radioactive Waste Disposal System* @Operation 000295
Solid Radioactive Waste Forms* 000266
Solid Radioactive Wastes* 000697
Solid Radioactive Wastes* the Organization of Sanitary 000567
Solid Waste Expected from a Large Commercial 000702
Solid Waste Forms* 000128
Solid Waste Storage Concepts* 000205
Solid Wastes from a Fuel Reprocessing Plant* 000097
Solidification and Burial Methods for Radioactive 000382
Solidification and Storage of Hanford's High-Level 000664
Solidification in Cements* @Immobilization of Strontium 000668
Solidification in Europe* 000611
Solidification of Fission Product Solutions Using a 000607
Solidification of Savannah River Plant High-Level Waste* 000616
Solidification of Savannah River Plant Waste* 000867
Solidification Agents used and Products Formed with 000621
Solidification Facility Off-Gas* of Technetium 99 000598
Solidification Method* @The Sandia 000636
Solidification Off-Gas* of Carbon 14, Krypton 85 000285
Solidification Process - A Broad Range Aqueous Waste 000636
Solidification System Resulting in an Extremely Stable 000597
Solidification Technology* 000580
Solidification* 000678
Solidification* @Characterization of 000122
Solidified High-Level Waste at Hanford* 000212
Solidified Radioactive Wastes* 000095
Solidified Simulated Reactor Waste Forms* 000115
Solidified Waste Characterization: The Scientific Basis 000002
Solidified Wastes in Underground Salt Mines* Salt 000316
Solids and Liquids in Salt Formations* 000350
Solids and Liquids in Salt Formations* 000360
Solids Storage Facility at Idaho Chemical Processing 000208
Solids Used in Shale Fracturing Grouts as Determined by 000681
Solids* @Application of Mass Transport 000258
Solution to the Long Time Storage Problem for High- 000428
Solution with Zeolite (2). Leachabilities of Cesium and 000619
Solution Mining: 1. Method of Analysis and Preliminary 000460
Solutions into a Stream Ecosystem* @A Transport 000311
Solutions of Nuclear Waste* 000486
Solutions Using a Drum Drier* @PIPS 000607
Solutions Using Bone Char - Pilot Study* 000578
Solutions* of the Appalachian Basin with Reference 000374
Solved Waste Problem* 000527
Solvent and on Fixation into Bitumen*Recent Experiments 000578
Solves Nuclear Waste Problem* 000159
Sorbents for Fixation in Concrete* 000655
Sorption and Fixation of Large Cations by Shale 000018
Sorption and Migration of Radionuclides in Geologic 000288
Sorption and Retention by Clay Minerals: Oak Ridge 000675
Sorption and Transport of Radionuclides in Hanford 000358
Sorption of Microquantities of Strontium and Cesium in 000085
Sorption of Technetium 99, Neptunium 237, and Americium 000298
Sorption on Two United States Subsoils from Differing 000293
Sorption Behavior of Cesium* 000033

PERMUTED TITLE INDEX

Applicability of Microautoradiography to
Studies of Strontium, Technetium, Cesium, Barium, 0
Processing Plant Fuel Storage Basin Water* Primary
Isolation Safety Assessment Program and Description of
Neutron and Gamma-Ray
Convection at Low Rayleigh Number from Concentrated
and Physical and Chemical Properties* 0
in the Southern Part of the Atlantic Coastal Plain,
Transuranic Waste at the Savannah River Plant, Aiken,
Contamination near a Transuranic Storage Area in
of the Virgin River Valley Salt Deposits, Clark County,
Report, Waste Isolation Pilot Plant (WIPP) Site,
Report, Waste Isolation Pilot Plant (WIPP) Site,
Report, Waste Isolation Pilot Plant (WIPP) Site,
Report, Waste Isolation Pilot Plant (WIPP) Site,
Report, Waste Isolation Pilot Plant (WIPP) Site,
Report, Waste Isolation Pilot Plant (WIPP) Site,
Report, Waste Isolation Pilot Plant (WIPP) Site,
Comparison of Plutonium Concentrations in Deer from the
Migration Studies Associated with the WIPP Site in
the Waste Storage Potential of Mesozoic Aquifers in the
Paleozoic Stratigraphy of Two Areas in
Handling of Radioactive Waste at
Suitable for Developing Dry Underground Storage
Executive Summary* Feasibility of
Technical Summary* Feasibility of
Importance of the Chemical
Cobalt 60 Leached from Matrix Effect of Dimension of
Raman
the Volatility of Cesium from Aluminosilicates by Raman
Consequences from Sabotage of Shipping Casks for
The Canadian Program for Storage and Disposal of
Evaluation of the Consequences of Light Water Reactor
Transportation of
Nuclear Power Industry* Projections of
Interim Storage of
Hot Cell Facilities* 0
Term Problems of High-Level and Actinide-Contaminated
the Transportation of Liquid Metal Fast Breeder Reactor
Fire Protection Device for LMFBR
Pre-Reactor Storage* 0
Tennessee Valley Authority Away-From-Reactor
Models for the Transportation of Radioactive Waste and
the Idaho Chemical Processing Plant Waste Canisters and
Level Reprocessing Waste. 4. Safety Handling of
Corrosion Resistance of Canisters for Final Disposal of
Retrofitting Trojan Nuclear Plant's
Experiments on the Treatment of Medium Level Wastes and
1. Method of Analysis and Preliminary Results for
Surface Layers on Simulated High-Level Waste Glasses by
Development of Glass Compositions for Immobilization of
History of Prototype High Level Waste Canister
Repository in a Bedded An Analysis of the Geological
Development and Radiation
0
Waste Solidification System Resulting in an Extremely
Fixation of Radioactivity in
Fixation of Radioactivity in
the Organization of Sanitary Radiometric Control at the
High Level Radioactive Waste Corrosion Evaluation of
Conductivity of the Rocks in the Bureau of Mines
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
0
0
Contribution to Draft Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Management Program at the National Reactor Testing
New Hungarian Radioactive Waste Handling and Disposal
and Problems at the National Reactor Testing
Hydrology of Waste Disposal National Reactor Testing
Handling of Radioactive Waste at Soviet Nuclear Power
Science Underlying Radioactive Waste Management:
Function and Licensing
Geologic Isolation of Radioactive Waste* The

Sorption Studies* 000003
Sorption-Desorption Studies on Granite. 1. Initial 000020
Source of Radionuclides Leaking into the Idaho Chemical 000188
Source Term Characterization Tank at Pacific Northwest 000315
Sources in Light Water Reactor High-Level Nuclear Waste* 000590
Sources in Porous Media* Steady Thermal 000037
Sources of Radioactive Waste from Light-Water Reactors 000693
South Carolina and Georgia* of Mesozoic Aquifers 000365
South Carolina* for Long-Term Management of Defense 000206
Southeastern Idaho* Plutonium and Americium 000181
Southeastern Nevada to Investigate their Suitability 000439
Southeastern New Mexico. Geochemistry* Characterization 000078
Southeastern New Mexico. Hydrology* Characterization 000456
Southeastern New Mexico. Regional Geology* 000453
Southeastern New Mexico. Seismology* Characterization 000455
Southeastern New Mexico. Site Geology* Characterization 000454
Southeastern New Mexico. Special Studies of Waste 000073
Southeastern New Mexico* Geological Characterization 000338
Southeastern United States and in Deer from an 000271
Southern New Mexico* Radionuclide 000240
Southern Part of the Atlantic Coastal Plain, South 000365
Southeastern Indiana* 000393
Soviet Nuclear Power Stations* 000628
Space in the Midcontinent, U.S.A.* Search for Carbonates 000371
Space Disposal of Radioactive Nuclear Waste. 1. 000438
Space Disposal of Radioactive Nuclear Waste. 2. 000433
Species Formed in the Interaction* 000032
Specimen on Amounts of Cesium 137, Strontium 90 and 000639
Spectra of Cesium Aluminosilicate Waste Forms* 000113
Spectrometry* Investigations of 000138
Spent Fuel and High-Level Waste* of Radiological 000586
Spent Fuel and High-Level Wastes* 000329
Spent Fuel and Plutonium Shipping Package Loss at Sea* 000118
Spent Fuel by Barge and By Rail: A Comparison of Risk* 000565
Spent Fuel to be Discharged by the United States 000690
Spent Fuel Assemblies* 000125
Spent Fuel Handling and Packaging Program. A Survey of 000135
Spent Fuel Reprocessing Wastes from the Uranium- 000035
Spent Fuel Rods* Evaluation of Organic Coolants for 000096
Spent Fuel Shipping Casks* 000568
Spent Fuel Storage Requirements - The Need for Away- 000705
Spent Fuel Storage Studies* 000320
Spent Fuel* Logistics 000566
Spent Fuels Preliminary Description and Operating 000138
Spent Nuclear Fuel and Final Storage of Vitrified High- 000123
Spent Nuclear Fuel* 0 000132
Spent Basin Transfer System* 000658
Spent Solvent and on Fixation into Bitumen* Recent 000578
Spherical Cavities* Cavities Formed by Solution Mining: 000580
Sputter-Induced Optical Emission* of Leached 000118
SRP Waste* 0 000656
SS-9 While in Air and Water Storage* 000101
Stability of a Hypothetical Radioactive Waste 000562
Stability of Glasses for Highly Radioactive Wastes* 00600
Stabilization of Simulated Commercial Waste Calcine* 000660
Stable Product* Rutile Microencapsulation: A Radioactive 000597
Stable, Solid Media* 000646
Stable, Solid Media* 000680
Stage of Transporting Solid Radioactive Wastes* is 000567
Stainless Steels in Idaho Chemical Processing Plant 000117
Standard Rock Salt* Thermal 000063
Statement (GEIS): Radioactive Waste in Geologic 000552
Statement (GEIS): Radioactive Waste Isolation in 000021
Statement (GEIS): Radioactive Waste Isolation in 000022
Statement (GEIS): Radioactive Waste Isolation in 000023
Statement (GEIS): Radioactive Waste Isolation in 000024
Statement (GEIS): Radioactive Waste Isolation in 000237
Statement (GEIS): Radioactive Waste Isolation in 000384
Statement (GEIS): Radioactive Waste Isolation in 000385
Statement (GEIS): Radioactive Waste Isolation in 000585
Statement (GEIS): Radioactive Waste Isolation in 000586
Statement (GEIS): Radioactive Waste Isolation in 000587
Statement (GEIS): Radioactive Waste Isolation in 000588
Statement (GEIS): Radioactive Waste Isolation in 000589
Statement of Dr. James E. Martin* 000514
Statement of Dr. Terry R. Lash* 000511
Statement on Commercial Waste Management: Radioactive 000457
Statement: Radioactive Waste Isolation in Geologic 000075
Statement: Radioactive Waste Isolation in Geologic 000296
Statement: Radioactive Waste Isolation in Geologic 000550
Station* High-Level Radioactive Waste 000671
Station* Technological Facilities at the 000593
Station* Operating Practices, Experiences, 000200
Station, Idaho, An Interim Report* 000410
Stations* 000628
Status and Needs* 000663
Status of the Asse Test Storage Facility* 000497
Status of Borehole Plugging and Shaft Sealing for 000536

PERMUTED TITLE INDEX

@New Mexico Waste Isolation Pilot Plant - A
Waste Management, Production, and Reprocessing @
Concentrated Sources in Porous Media@ @
Radioactive Waste @Corrosion Evaluation of Stainless
@Water Experiments in the Climax
@ @The Canadian Program for
Radioactive Waste Repositories@ @
from Nuclear Facilities and General @Engineered
American Cooperative Program on Radioactive Waste
Waste Management. Volume 2. Implementation for Terminal
@Feasibility Study of Underground
and Temperature Effects in Case of Underground
@The Management and
Wastes@ @Solidification and
@Retrievable Surface
Formations@ @Ultimate
Formations@ @Ultimate
@Clays as the First Barrier for Long-Term
Idaho National @Transport, Handling, and Interim
Scientific Basis for, and Experience With, Underground
@Radiolytic Gas Production During Long-Term
Products and Applications to the Problem of Geological
Rocks@ @An Evaluation of the
Nevada to Investigate their Suitability for Possible
Basin to Investigate Their Suitability for Possible
@Secure
@Terminal
Texas Panhandle and Western Oklahoma for Underground
@Interim
Safety @Handling of Spent Nuclear Fuel and Final
of the Utility of Gulf Coast Salt Domes for the
and American Contamination near a Transuranic
@Removal of Cesium and Strontium from Fuel
Looking into the Idaho Chemical Processing Plant Pool
@Leak Testing Fuel Stored in the ICPP Fuel
@Corrosion Monitoring of
of Allied-General Nuclear Services On-Site Solid Waste
@Underground
@Above-Ground
Alpha Repository: 2. Conceptual Layouts of Underground
Calcine Retrieved from the Second Calcined Solids
@Function and Licensing Status of the Asse Test
Management@ @The Development of Process and
@Mathematical Model of Waste
Aquifer System Below the @Geologic Evaluation of Waste-
Geologic and Hydrologic Factors Related to the Waste
@A Possible Solution to the Long Time
1976, to September 30, 1977@ @National Waste Terminal
@National Waste Terminal
@Development of National Waste Terminal
Site Selection Factors For the National Waste Terminal
(AIRM) Studies Relative to the National Waste Terminal
1. Management Summary@ @National Waste Terminal
@Dissolubility Study of the Terminal Waste
Transportation System@ @National Waste Terminal
Storage@ @Spent Fuel
@National Waste Terminal
@Gold and Selective
for Carbonates Suitable for Developing Dry Underground
Tennessee Valley Authority Away-From-Reactor Spent Fuel
Pipe Cooling System for Underground, Radioactive Waste
of Cooling Coil Corrosion in Radioactive-Waste
Isolation Pilot Plant (@Studies of Transuranic Waste
@Considerations Relevant to Provisional Versus Final
Storage Requirements - The Need for Away-From-Reactor
High Level Waste Canister 52-9 While in Air and Water
of Wastes and Containers During Retrievable Geologic
@Leak Testing Fuel
Chemical Processing Plant@ @Sampling of
Evaluations of Alternatives for Long-Term Management of
on Accumulation of Actinide Critical Masses from
@The Feasibility of
@Durability of Containers for
Analysis of Shipping Containers@An Assessment of Stress-
Potential Repository Rocks@ @Stress,
Disposal in Geologic Basins - A Study of Reservoir
Industrial Potential of Geologic Basins and Reservoir
; Radioactive Waste Isolation in Geologic Formations.
@Paleozoic
Assessment of Plutonium 239, 240 Concentrations in a
Zero Release of Low-Level Radionuclide Solutions into a
Continuous Tritium Monitor for Nuclear Facility Off-Gas
for the Removal of Uranium and Plutonium from Waste
of Waste-Container Sleeve: 3. Influence of Salt
@An Analysis of the Measured Values for the State of
in Geologic Formations. Volume 20. Thermomechanical
Plastic Analysis of Shipping @An Assessment of
@Effects of Temperature, Temperature Gradients,
Status Report@ 000082
Status Report, Waste Incineration and Fixation for 000090
Steady Thermal Convection at Low Rayleigh Number from 000137
Steels in Idaho Chemical Processing Plant High Level 000117
Stock, Nevada Test Site@ 000046
Storage and Disposal of Spent Fuel and High-Level Wastes 000329
Storage and Release of Radiation Energy in Salt in 000042
Storage for the Disposal of Non-Fuel Bearing Components 000098
Storage in Mined Caverns in Crystalline Rock, Program 000407
Storage in Reference Repository and Other Applications@ 000279
Storage of Bitumen Blocks Containing Radioactive Wastes@ 000501
Storage of Bitumen@ @Radiolysis 000202
Storage of Commercial Power Reactor Wastes@ 000601
Storage of Hanford's High-Level Radioactive Liquid 000649
Storage of High-Level Radioactive Wastes@ 000158
Storage of High-Level Waste Solids and Liquids in Salt 000359
Storage of High-Level Waste Solids and Liquids in Salt 000360
Storage of Highly Active Radioactive Wastes@ 000128
Storage of Intermediate-Level Transuranic Waste at the 000187
Storage of Liquid Radioactive Wastes in the USSR@ @The 000464
Storage of Nuclear Wastes@ 000577
Storage of Nuclear Wastes@ Paths for Ohio Reactor 000236
Storage of Radioactive Waste within Silicic Pyroclastic 000077
Storage of Radioactive Waste Material as of September, 000039
Storage of Radioactive Waste Material@ Texas Salt Dome 000030
Storage of Radioactive Waste@ 000587
Storage of Radioactive Wastes in Geologic Formations@ 000025
Storage of Radioactive Wastes@ Salt Deposits in the 000017
Storage of Spent Fuel Assemblies@ 000125
Storage of Vitrified High-Level Reprocessing Waste. 4. 000123
Storage or Disposal of Radioactive Wastes@Investigation 000020
Storage Area in Southeastern Idaho@ @Plutonium 000181
Storage Basin Water@ 000687
Storage Basin Water@ @Primary Source of Radionuclides 000184
Storage Basin@ 000631
Storage Bins for Radioactive Calcines@ 000116
Storage Concepts@ @Description 000205
Storage Condition for Radioactive Wastes@ 000085
Storage Facilities for Vitrified Highly Active Wastes@ 000573
Storage Facilities@ @Design Aspects of the 000541
Storage Facility at Idaho Chemical Processing Plant@ 000204
Storage Facility@ 000497
Storage Materials Suitable for Krypton 85 Waste 000181
Storage Operations@ 000667
Storage Potential in Selected Segments of the Mesozoic 000366
Storage Potential of Mesozoic Aquifers in the Southern 000365
Storage Problem for High-Level Waste@ 000424
Storage Program Progress Report for Period October 1, 000313
Storage Program Progress Report June 1977@ 000312
Storage Program Waste Packaging Criteria@ 000127
Storage Program@ @Preliminary Geologic 000488
Storage Program@ @As Low as is Reasonably Achievable 000493
Storage Program, Configuration Management Plan. Volume 000499
Storage Program, Plume Investigation@ 000396
Storage Program: Potential Problems in the Waste 000568
Storage Requirements - The Need for Away-From-Reactor 000705
Storage Review Meeting on Salt Dissolution@ 000348
Storage Solves Nuclear Waste Problem@ 000159
Storage Space in the Midcontinent, U.S.A.@ Search 000371
Storage Studies@ @ 000320
Storage Tanks@ @Heat 000107
Storage Tanks@ @Investigation 000139
Storage Under Conditions Expected in the Waste 000047
Storage@ 000559
Storage@ @Spent Fuel 000705
Storage@ @History of Prototype 000101
Storage@ @Maximum Acceptable Temperatures 000121
Stored in the ICPP Fuel Storage Basin@ 000631
Stored High-Level Radioactive Calcined Waste at Idaho 000154
Stored Idaho National Engineering Laboratory 000380
Stored Nuclear Waste in Natural Rock Repositories@ 000015
Storing Radioactive Wastes in Columbia River Basalts@ 000606
Storing Solidified Radioactive Wastes@ 000095
Strain Data Suitable for Finite-Element Elastic-Plastic 000183
Strain, and Temperature Induced Permeability Changes in 000036
Strata@ @Subsurface 000000
Strata@ @Economic and 000399
Stratigraphics of Salt, Granite, Shale, and Basalt@(EIS) 000385
Stratigraphy of Two Areas in Southwestern Indiana@ 000393
Stream near Argonne National Laboratory@ @A Preliminary 000201
Stream Ecosystem@ @A Transport Model for Long 000311
Streams@ @A 000172
Streams@ @Evaluation of New Macroporous Resins 000623
Strength on Sleeve Loading@ Plastic Analysis 000040
Stress in the Earth's Crust@ 000414
Stress Analysis and Development of Thermal Loading 000028
Stress-Strain Data Suitable for Finite-Element Elastic- 000183
Stress, and Irradiation on Migration of Urine 000041

PERMUTED TITLE INDEX

Changes in *-'ential Repository Rocks* 0 Stress, Strain, and Temperature Induced Permeability 000036
Rock Stresses Associated with Burial of Nuclear Waste 000085
The Mechanical Properties of Stripa Granite 000469
A Pilot Heater Test in the Stripa Granite 000535
and Isotope Hydrology of Groundwaters in the Stripa Granite: Results and Preliminary Interpretation* 000397
Theoretical Temperature Fields for the Stripa Heater Project, Volume 1 000374
Borehole Drilling and Related Activities at the Stripa Mine 000422
Electrical Heaters for Thermo-Mechanical Tests at the Stripa Mine 000538
Used in the Underground Tunnels and Test Rooms at Stripa* *Mining Methods 000350
Handling, and Display for the Heater Experiments at Stripa* *Data Acquisition, 000056
on Geophysical and Mechanical Borehole Measurements at Stripa* *Preliminary Report 000437
Full-Scale and Time-Scale Heating Experiments at Stripa: Preliminary Results 000391
An Approach to the Fracture Hydrology at Stripa: Preliminary Results 000390
Wastes by Solidification in Cements* *Immobilization of Strontium and Cesium in Intermediate-Level Liquid 000668
The Sorption of Microquantities of Strontium and Cesium in Soils 000045
Solution with Zeolite (Z). Leachabilities of Cesium and Strontium from Calcined Zeolites* of Radioactive Waste 000619
Removal of Cesium and Strontium from Fuel Storage Basin Water 000687
Texture* *Migration of Strontium 90 and Cerium 144 in Soils of Different 000227
of Dimension of Specimen on Amounts of Cesium 137, Strontium 90 and Cobalt 60 Leached from Matrix of 000639
in Aerated Zones. 3. Theoretical Estimates of Strontium 90 Migration* *Radionuclides Migration 000288
Montmorillonite Exchange Equilibria with Strontium-Sodium-Cesium 000027
Plant Waste Supernate* *A Study of Methods for Removing Strontium, Plutonium, and Ruthenium from Savannah River 000688
Desorption Studies on Granite. 1. Initial Studies of Strontium, Technetium, Cesium, Barium, Cerium, 000028
Mining: 1. Method of Analysis and Preliminary Results 0 Structural Analysis of Salt Cavities Formed by Solution 000580
Possibilities for Subsurface Waste Disposal in a Structural Syncline in Pennsylvania 000408
The Utilization of the NUL Data Structure in Consequence Analysis for WIPP Repository 000223
237, and Technetium 99 Sorption on Two United States Subsoils from Differing Weathering Intensity Areas* 000293
Technetium 99, Neptunium 237, and Americium 241 on Two Subsoils from Differing Weathering Intensity Areas* of 000294
Sorption and Transport of Radionuclides in Hanford Subsoils* *Variables Affecting 000358
Test Holes in Central Kansas, Part 1: Salt Beds in the Subsurface in Russell, Lincoln, Ellsworth, Barton, and 000879
Reservoir Strata* 0 Subsurface Disposal in Geologic Basins - A Study of 000800
Well Injection* 0 Subsurface Disposal of Liquid Industrial Wastes by Deep- 000400
Summary of the Appalachian Basin with Reference to the Subsurface Disposal of Radioactive Waste Solutions* 000379
Site, Salt Valley Anticline, Grand County, Utah* 0 Subsurface Geology of a Potential Waste Emplacement 000411
New *Possibilities for Disposal of Industrial Wastes in Subsurface Rocks on North Flank of Appalachian Basin in 000429
Pennsylvania* *Possibilities for Subsurface Waste Disposal in a Structural Syncline in 000408
Geology of Subsurface Waste Disposal in Michigan Basin 000364
Kansas* 0 Subsurface Waste-Disposal Potential in Salina Basin of 000394
of Radionuclides* 0 Suggested Concentration Limits for Shallow Land Burial 000277
of Northeast Texas Salt Dome Basin to Investigate Their Suitability for Possible Storage of Radioactive Waste 000438
Clark County, Southeastern Nevada to Investigate their Suitability for Possible Storage of Radioactive Waste 000439
of Anhydrite and Brine and Their Implications for the Suitability of Anhydrite as a Nuclear Waste Repository 000016
of the Rocks in the Bureau of Mines Standard Rock Suite* *Thermal Conductivity 000063
Defense High-Level Radioactive Waste* 0 Summary - Alternatives for Long-Term Management of 000478
Subsurface Disposal of Radioactive Waste *Geologic Summary of the Appalachian Basin with Reference to the 000379
0 Summary of Previous Pertinent Studies* 000013
Nuclear Power Plants from 1973 thru 1976* 0 Summary of Radioactivity Released in Effluents from 000704
Radioactive Waste Disposal* 0 Summary Review of Rock Mechanics Workshop on 000373
Disposal of Radioactive Nuclear Waste. 2. Technical Summary* *Feasibility of Space 000833
Disposal of Radioactive Nuclear Waste. 1. Executive Summary* *Feasibility of Space 000838
Waste in Geologic Formations. Volume 1. Executive Summary* Impact Statement (GEIS): Radioactive 000552
Storage in Mined Caverns in Crystalline Rock, Program Summary* Cooperative Program on Radioactive Waste 000887
Configuration Management Plan. Volume 1. Management Summary* *National Waste Terminal Storage Program, 000499
Transportation of Radioactive Waste-Materials to the Sea 000468
and Ruthenium from Savannah River Plant Waste Supernate*of Methods for Removing Strontium, Plutonium, 000689
Serpentinite Bodies from the Appalachian Mountain 0 Supplementary Report: Geologic Feasibility of Talc and 000484
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000021
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000022
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000023
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000024
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000237
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000384
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000385
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000545
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000546
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000547
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000548
GEIS): Radioactive Waste Isolation in *Technical Support for Generic Environmental Impact Statement (000549
GEIS): Radioactive Waste in Geologic *Technical Support for Generic Environmental Impact Statement (000552
Radioactive Waste Isolation in Geologic *Technical Support for Generic Environmental Impact Statement: 000075
Radioactive Waste Isolation in Geologic *Technical Support for Generic Environmental Impact Statement: 000296
Radioactive Waste Isolation in Geologic *Technical Support for Generic Environmental Impact Statement: 000550
A Review of the Heat Surface Heater Experiment at Oak Ridge, Tennessee 000421
Sputter-Induced Optical *Characterization of Leached Surface Layers on Simulated High-Level Waste Glasses by 000119
Retrievable Surface Storage of High-Level Radioactive Wastes 000158
Simulation of Sediment and Contaminant Transport in Surface Waters* 1977 to September 1978 Mathematical 000289
Unsaturated Soils after Water Disposal into a Limited Surface, Part 2* Soils: Behavior of Radioactive Ions in 000292
of Radiations from Transuranium Nuclides on Container Surfaces* *Effects 000133
Influence of Surfactants on Mitigation of Radioactive Waste 000626
Interactions Between Nuclear Waste and Surrounding Rock 000282
In Situ Test Facilities* 0 Survey of Active and Inactive Mines for Possible Use as 000378
Spent Fuel Handling and Packaging Program. A Survey of Hot Cell Facilities 000135
Deep Geologic Formations: A Perspective on the 0 Survey of Naturally Occurring Hazardous Materials in 000088
Salt Repositories* *United States Geological Survey Studies of Physical-Chemical Relationships in 000079
Ion Exchange Research by the United States Geological Survey 000064
Adsorption Studies by the U.S. Geological Survey 000576
Nuclear Power in Sweden 000322
Nuclide Transport by Groundwater in Swedish Bedrock 000001

PERMUTED TITLE INDEX

Waste Storage in Mined Caverns in Crystalline Rock, $ 000687
$International Waste-Management
for Subsurface Waste Disposal in a Structural 000522
$Incorporation of High-Level Radwaste in 000809
Waste for Geologic Disposal* $ 000185
$Radiation Damage Studies on Natural and 000820
Electron $Radiation Damage Studies on Natural and 000817
Waste Solidification System Resulting in an Extremely $ 000082
Canisters and Spent Fuels $Neutron Interrogator Assay 000587
$An Automatic Drum Scanning 000136
Nuclear Fuel Cycle: January-March 1978$Volume Reduction 000706
Nuclear Fuel Cycle: July-September $Volume Reduction 000570
Fuel Cycle: October-December 1977* $Volume Reduction 000571
July-December 1977* $Fluidized Bed Incineration 000628
$Heat Pipe Cooling 000580
Distribution in a Desert Pavement-Desert Wound Soil 000107
Potential in Selected Segments of the Mesozoic Aquifer 000305
Wastes at Hanford* $ 000366
$Influence of Radiation on the 000202
Microencapsulation: A Radioactive Waste Solidification 000078
Republic of Germany. Vol 8. The Law Concerning the $ 000487
Trojan Nuclear Plant's Spent Resin Transfer 000495
Program: Potential Problems in the Waste Transportation 000658
of Radiostrontium in Sacley Soil. Three-Component 000565
a Function of pH and Certain Soil Components: NPO3-RADE 000288
Plant Liquid and Solid Radioactive Waste Disposal 000566
Techniques for $Preliminary Concept of Waste Retrieval 000630
$TRU Waste Cyclone Drum Incinerator and Treatment 000622
Heads* $The Design of 000670
$Deep Disposal 000255
$Radioactive Waste Transportation 000187
$ 000505
$Alternative Waste Management 000328
Implications for the Suitability of Anhydrite as a $P- 000016
$High-Level Waste Leakage from the 241- 000198
$241- 000162
Highly Radioactive Glass Blocks Buried Below the Water 000685
Mountain $Supplementary Report: Geologic Feasibility of 000488
$High-Level Waste Leakage from the 241-T-106 000198
$Geology of the 241-AY 000193
$Geology of the 241-B 000198
$Geology of the 241-BX 000195
$241-T-106 000162
for $Preliminary Concept of Waste Retrieval System- 000630
of Cooling Coil Corrosion in Radioactive-Waste Storage 000139
System for Underground, Radioactive Waste Storage 000107
Program and Description of Source Term Characterization 000315
Unweathered Igneous Rocks* $Chemistry of 000228
$Geologic Migration Potentials of 000033
Evaluation and Development for the Monitoring of 000598
from Differing $Americium 241, Neptunium 237, and 000293
for the Determination of Environmental Iodine 129 and 000008
Subsoils from Differing Weathering $Sorption of 000298
Studies on Granite. 1. Initial Studies of Strontium, 000028
of Cores from Holes P-3 HH-1 through P-3 HH-5, Area G, 000196
Defense Waste* $ 000627
Radioactive Waste* $Earth Science 000503
of Space Disposal of Radioactive Nuclear Waste. 2. 000822
Statement (GEIS): Radioactive Waste Isolation in $ 000021
Statement (GEIS): Radioactive Waste Isolation in $ 000022
Statement (GEIS): Radioactive Waste Isolation in $ 000023
Statement (GEIS): Radioactive Waste Isolation in $ 000026
Statement: Radioactive Waste Isolation in Geologic $ 000075
Statement (GEIS): Radioactive Waste Isolation in $ 000227
Statement: Radioactive Waste Isolation in Geologic $ 000296
Statement (GEIS): Radioactive Waste Isolation in $ 000384
Statement (GEIS): Radioactive Waste Isolation in $ 000285
Statement (GEIS): Radioactive Waste Isolation in $ 000545
Statement (GEIS): Radioactive Waste Isolation in $ 000546
Statement (GEIS): Radioactive Waste Isolation in $ 000547
Statement (GEIS): Radioactive Waste Isolation in $ 000548
Statement (GEIS): Radioactive Waste Isolation in $ 000549
Statement: Radioactive Waste Isolation in Geologic $ 000550
Statement (GEIS): Radioactive Waste in Geologic $ 000552
Radioactive Waste Handling and Disposal Station* $ 000593
of the Energy Research and Development Administration 000329
$Rockwell Hanford Operations Quarterly Report, Process 000225
Repository Sites: Progress Report for the $Cement 000059
Methods Applied to Radioactive Waste Management* $ 000603
$Current High-Level Waste Solidification 000580
Base for Waste Repository Design, Volume 6. Excavation 000583
$Experimental Permeability Studies at Elevated 000069
Physical Chemistry of Fluids in Rock Salt at Elevated 000080
Waste Fixation in PUREX (Formed Under Elevated 000060
Bitumen* $Radiolysis and 000202
Volume 1* $Theoretical 000378
in the Nuclear Fuel Cycle - Reprocessing of High- 000501
Migration of Brine Inclusions $Effects of Temperature, 000041
Swedish-American Cooperative Program on Radioactive
Symposium*
Syncline in Pennsylvania* $Possibilities
SYMBOC*
Synthesis of Nuclear Waste Monazites, Ideal Actinide
Synthetic Rock Salt for Waste Disposal Applications*
Synthetic Rock Salt Utilizing Measurements Made During
Synthetic Ductile Microencapsulation: A Radioactive
System for the Idaho Chemical Processing Plant Waste
System for the Measurement of Plutonium in Waste*
System for Solid and Liquid Transuranic Waste from the
System for Solid and Liquid Transuranic Waste from the
System for Solid and Liquid TRU Waste from the Nuclear
System for U.S. Department of Energy Defense Waste,
System for Underground, Radioactive Waste Storage Tanks*
System in Area 11* $Plutonium
System Below the Zone of Fresh Water, Atlantic Coastal
System Design for Retrieval of Solidified High-Level
System Liquid Radioactive Wastes - Geologic Formation*
System Resulting in an Extremely Stable Product* Ductile
System Study, Radioactive Wastes in the Federal
System* $Retrofitting
System* $National Waste Terminal Storage
System* $Preliminary Investigations of the Adsorption
System* 238 and Curium 242 from a Contaminated Soil as
System* $Operation of the Monticello Nuclear Generating
System-Tank Wall Access Using Underground Mining
System: July-September, 1978*
Systems for Handling Radioactive Ion Exchange Resin
Systems for Radioactive Wastes*
Systems Analysis and Program Plan*
Systems Analysis Study for Waste Management Criteria*
Systems*
T-I Relations of Anhydrite and Brine and Their
T-106 Tank at Hanford*
T-106 Tank Leak Investigation*
Table: Fifteen Years of Results* of Radioactivity from
Talc and Serpentinite Bodies from the Appalachian
Tank at Hanford*
Tank Farm*
Tank Farm*
Tank Farm*
Tank Leak Investigation*
Tank Wall Access Using Underground Mining Techniques
Tanks* $Investigation
Tanks* $Heat Pipe Cooling
Task at Pacific Northwest Laboratories* Assessment
Technetium 99 and Neptunium 237 in Contact with
Technetium 99 and Neptunium 237*
Technetium 99 and Selenium 79 in Nuclear Fuel
Technetium 99 Sorption on Two United States Subsoils
Technetium 99* $Methodology
Technetium 99, Neptunium 237, and Americium 241 on Two
Technetium, Cesium, Barium, Cerium, Europium, Uranium,
Technical Area 58* $Geologic Description
Technical Challenges in the Immobilization of Hanford
Technical Plan for Mined Geologic Disposal of
Technical Summary* $Feasibility
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technical Support for Generic Environmental Impact
Technological Facilities at the New Hungarian
Technological Program for Radioactive Waste Management
Technology and Process Development, July 1977 through
Technology for Plugging Boreholes in Radioactive Waste
Technology Assessment and Impact Analysis of Separation
Technology*
Technology* $Information
Temperature and Pressure of Granitic Rocks*
Temperature and Pressure to Repositories for
Temperature and Pressure) Concretes - Experimental
Temperature Effects in Case of Underground Storage of
Temperature Fields for the Strips Heater Project,
Temperature Gas-Cooled Reactor Fuel Containing Uranium
Temperature Gradients, Stress, and Irradiation on

Repository Rocks* $Stress, Strain, and
$Long-Term Elevated
Irradiation on Migration of Brine $Effects of
Retrievable Geologic Storage* $Maximum Acceptable
Element Transport in Lithic Material by Fluid at High
and Yazoo Clay in Parts of Alabama, Mississippi and
Storage Studies* $
of the Near Surface Heater Experiment at Oak Ridge,
Safety Assessment Program and Description of Source
$Long-
Crystalline Rock Repository Media* $Long
$Effect of Actinide Removal on the Long-
Grained Deposits* $A Concept for Long-
Idaho Chemical Processing Plant, $Alternative for Long-
Hanford Reservations, Richland, $Alternatives for Long-
$Summary - Alternatives for Long-
Savannah River Plant, Aiken, $Alternatives for Long-
and Other Evaluations of Alternatives for Long-
Fuel Reprocessing Plants* $Long-
Spent Fuel Reprocessing Wastes $Some Aspects and Long-
Stream Ecosystem* $A Transport Model for Long
Geologic Formations* $Long-
Repositories* $Long-
$Glass as the First Barrier for Long-
$Radiolytic Gas Production during Long-
Waste Management. Volume 2. Implementation for
Formations* $
October 1, 1976, to September 30, 1977* $National Waste
$National Waste
$Development of National Waste
Geologic Site Selection Factors For the National Waste
(NIAPP) Studies Relative to the National Waste
Plan. Volume 1. Management Summary* $National Waste
Waste Transportation System* $National Waste
$National Waste
$Mineability Study of the
$Statement of Dr.
Storage of $Evaluation of Permian Salt Deposits in the
$Geologic Study of the Interior Salt Domes of Northeast
$Regional Geology of the Northeast
$Genesis and Growth of Salt Domes in the Northeast
$Geology of Certain Domes of the Northeast
of Strontium 90 and Cerium 144 in Soils of Different
Transport in Selected Rocks: 1977 Annual Report of $
$Radionuclides Migration in Aerated Zones. 3.
Project, Volume 1* $
Random-Walk $Multicomponent Mass Transport Model:
Solids* $Application of Mass Transport
Waste Isolation in Geologic Formations. Volume 19 -
$
Mines Standard Rock Suite* $
Dores* $
Concentrated Sources in Porous Media* $Steady
Repositories in Hard Rock. Parts 1 and 2$Mechanical and
$
$Geothermometry Methods for Determining the
20. Thermomechanical Stress Analysis and Development of
Hanford Nuclear Reservation* $
$Nuclear Waste Canister
$Electrical Heaters for
of Radionuclides in Geomedia: Oklo and Other Examples* $
Sleeve: 1. Initial Estimates of Loading on the Sleeve* $
Sleeve: 2. Air Gap Influence on Hole Closure* $
Sleeve: 3. Influence of Salt Strength on Sleeve Loading$
Waste Isolation in Geologic Formations. Volume 20.
$
$Shipments of Nuclear Fuel and Waste...Are
Liquid Radioactive Wastes Using Mechanically Agitated
of Uranium Process Wastes from the Uranium 233-
Gas-Cooled Reactor Fuel Containing Uranium 233 and
Fuel Reprocessing Wastes from the Uranium-Plutonium and
in Effluents from Nuclear Power Plants from 1973
Level Nuclear Waste Management in the United States: A
$A Possible Solution to the Long
Results* $Full-Scale and
Solidification* $Characterization of a
Research on Waste Disposal Off the Coast of
High Temperatures* $
$Analysis of Some
$Improved Heat
Multivalent Elements by Geologic Media* $Electron
$Retrofitting Trojan Nuclear Plant's Spent Basin
Argillite from Area 17, Nevada Test Site, Using the
$Review of Criteria for Packaging Plutonium for
$Nuclide
Temperatures* $Trace-Element
$Theoretical and Experimental Evaluation of Waste
Mathematical Simulation of Sediment and Contaminant

Temperature Induced Permeability Changes in Potential 000036
Temperature Leaching of Solid Waste Forms* 000128
Temperature, Temperature Gradients, Stress, and 000041
Temperatures of Wastes and Containers During 000121
Temperatures* $Trace- 000297
Tennessee as Potential Host Rocks for Underground 000630
Tennessee Valley Authority Away-From-Reactor Spent Fuel 000320
Tennessee* $A Review 000421
Term Characterization Task at Pacific Northwest 000315
Term Elevated Temperature Leaching of Solid Waste Forms* 000128
Term Geochemical Interactions of High Active Waste with 000157
Term Hazard of High-Level Waste* 000585
Term Isolation of Solid Radioactive Waste in Fine- 000376
Term Management of Defense High-Level Radioactive Waste 000346
Term Management of Defense High-Level Radioactive Waste 000347
Term Management of Defense High-Level Radioactive Waste* 000474
Term Management of Defense Transuranic Waste at the 000206
Term Management of Stored Idaho National Engineering 000380
Term Planning for Management of Aqueous Wastes from 000523
Term Problems of High-Level and Actinide-Contaminated 000035
Term Release of Low-Level Radionuclide Solutions into a 000311
Term Risk Assessment of Radioactive Waste Disposal in 000256
Term Safety Assessment of Geologic Isolation 000369
Term Storage of Highly Active Radioactive Wastes* 000128
Term Storage of Nuclear Wastes* 000577
Terminal Storage in Reference Repository and Other 000279
Terminal Storage of Radioactive Wastes in Geologic 000425
Terminal Storage Program Progress Report for Period 000313
Terminal Storage Program Progress Report June 1977* 000312
Terminal Storage Program Waste Packaging Criteria* 000127
Terminal Storage Program* $Preliminary 000888
Terminal Storage Program* as is Reasonably Achievable 000493
Terminal Storage Program, Configuration Management 000499
Terminal Storage Program: Potential Problems in the 000568
Terminal Storage Review Meeting on Salt Dissolution* 000348
Terminal Waste Storage Program, Eleana Investigation* 000396
Terry R. Lash* 000511
Texas Panhandle and Western Oklahoma for Underground 000417
Texas Salt Dome Basin to Investigate Their Suitability 000438
Texas Salt Dome Basin* 000412
Texas Salt Dome Basin* 000432
Texas Salt-Dome Basin* 000353
Texture* $Migration 000227
Theoretical and Experimental Evaluation of Waste 000219
Theoretical Estimation of Strontium 90 Migration* 000288
Theoretical Temperature Fields for the Strips Heater 000378
Theory and Numerical Implementation (Discrete-Parcel- 000215
Theory to the Leaching of Radionuclides from Waste 000258
Thermal Analyses* Impact Statement: Radioactive 000075
Thermal Calculations in Shale* 000562
Thermal Conductivity of the Rocks in the Bureau of 000063
Thermal Conductivity of Rock Salt from Louisiana Salt 000061
Thermal Convection at Low Rayleigh Number from 000037
Thermal Design Considerations for Radioactive Waste 000019
Thermal Effects in Shales: Measurements and Modeling* 000057
Thermal History of Shales* 000087
Thermal Loading Guidelines* Geologic Formations. Volume 000028
Thermal Properties of Gable Mountain Basalt Cores 000053
Thermally Induced Motion* 000025
Thermo-Mechanical Tests at the Strips Mine* 000538
Thermodynamic Considerations Underlying the Migration 000231
Thermoelastic/Plastic Analysis of Waste-Container 000067
Thermoelastic/Plastic Analysis of Waste-Container 000140
Thermoelastic/Plastic Analysis of Waste-Container 000170
Thermomechanical Stress Analysis and Development of 000028
Thermophysical Properties of Conasauga Shale* 000076
They Really Safe?* 000152
Thin-Film Evaporators* $Volume Reduction of 000592
Thorium 232 Cycle* $Migration 000251
Thorium* Fuel Cycle - Reprocessing of High-Temperature 000501
Thorium-Uranium Fuel Cycles* Actinide-Contaminated Spent 000035
thru 1976* $Summary of Radioactivity Released 000708
time for Decisions* $High- 000513
Time Storage Problem for High-Level Waste* 000424
Time-Scale Heating Experiments at Strips: Preliminary 000381
Titanate Based Ceramic for High Level Nuclear Waste 000422
Tokai-Mura* $Oceanographic 000415
Trace-Element Transport in Lithic Material by Fluid at 000297
Tracer Runs in Granite Rock Using a Pissure Model* 000287
Transfer from Radioactive Waste Canisters* 000119
Transfer Method of Controlling Eh During Adsorption of 000070
Transfer System* 000658
Transient Method* $Measuring the Permeability of Eleana 000088
Transport by Air* 000092
Transport by Groundwater in Swedish Bedrock* 000001
Transport in Lithic Material by Fluid at High 000297
Transport in Selected Rocks: 1977 Annual Report of 000219
Transport in Surface Waters* 1977 to September 1978 000289

Fresh Water Aquifer. 1. @Investigation of Salt
Fresh Water Aquifer. 2. @Investigation of Salt
Fresh Water Aquifer. 3. @Investigation of Salt
@Variables Affecting Sorption and
Galerkin Model* @Material
Assessment Program - Collection and Generation of
Incidents* @Hanford Groundwater
Radionuclide Solutions into a Stream Ecosystem* @A
Discrete-Parcel-Random-Walk @Multicomponent Mass
@
Waste Solids* @Application of Mass
Volume 21. Ground Water Movement and Nuclide
Intermediate-Level Transuranic Waste at the Idaho @
Spent Fuel Rods*@Evaluation of Organic Coolants for the
@Logistics Models for the
@ @
Comparison of Risk* @
@Release of Radionuclides from High-Level Waste During
June 1976* @Nuclear Waste Management and
Through December 1975* @Nuclear Waste Management and
Storage Program: Potential Problems in the Waste
@Radioactive Waste
Hardware @Experience in Volume Reduction, Packaging,
of Sanitary Radionuclide Control at the Stage of
of Microbial Processes on the Fate and Behavior of
@Plutonium and American Contamination near a
Laboratory* @Retrieval of Buried
Handling, and Interim Storage of Intermediate-Level
South @Alternatives for Long-Term Management of Defense
March 1978@Volume Reduction System for Solid and Liquid
September @Volume Reduction System for Solid and Liquid
@Selection of a Criticality Monitoring Technique for a
Public* @Effects of Current
@Environmental Effects from Current Operations of
Alternatives for Idaho National Engineering Laboratory
the Waste Isolation Pilot Plant (WIPP), @Studies of
@Environmental Effects of Alternatives for Managing
of Stored Idaho National Engineering Laboratory
Waste Isolation Pilot @Potential Microbial Impact on
Waste Isolation Pilot @Potential Microbial Impact on
Waste Isolation Pilot @Potential Microbial Impact on
@Potential Microbial Impact on
@Gas Generation from Radiolytic Attack of
@Determination of
@
@Effects of Radiations from
Installations* @Studies on
Effluents in the @Correlation of Radioactive Waste
@Geotechnical Analysis of Soil Samples from Test
Possibility of Disposing of Radioactive Waste in Ocean
@Mechanical Behavior of New Mexico Rock Salt in
@Radiolytic Gas Production from
@A Continuous
@Linde 13X Equilibrium with
@Retrofitting
1977* @Volume Reduction System for Solid and Liquid
Spaces: July-September, 1978* @
Nevada* @Petrology of
@Mining Methods Used in the Underground
@Nuclear Waste:
Water Characteristics of the Hanford Reservation of the
1977* @Fluidized Bed Incineration System for
@Adsorption Studies by the
@Risk Assessment Review Group Report to the
Nuclear Wastes* @
Dry Underground Storage Space in the Midcontinent,
An Important Safety Aspect for their Exploitation for
Processing of High-Level Atomic Wastes with the View to
in Salt Formations* @
in Salt Formations* @
@Fracture Detection in Crystalline Rock Using
on the Accumulation of Radionuclides by Green Algae (
@
Mississippi and Tennessee as Potential Host Rocks for
Environmental Evaluation of Nuclear Waste Disposal by
@Report of Investigation
of Waste Retrieval System-Tank Wall Access Using
and Crystalline Rocks: Some @An Appraisal of
@An Appraisal of Hard Rock for Potential
@An Appraisal of Hard Rock for Potential
of the Disposal of High Activity Solidified Wastes in
Radioactive Wastes* @Feasibility Study of
@Radiolysis and Temperature Effects in Case of
USSR* @The Scientific Basis for, and Experience With,
in the Texas Panhandle and Western Oklahoma for
@
of the Alpha Repository: 2. Conceptual Layouts of
Search for Carbonates Suitable for Developing Dry

Transport in Vertical Boreholes and Brine Invasion into 000273
Transport in Vertical Boreholes and Brine Invasion into 000274
Transport in Vertical Boreholes and Brine Invasion into 000275
Transport of Radionuclides in Hanford Subsoils* 000358
Transport through Porous Media: A Finite-Element 000243
Transport Data* No. 8590 UK. Waste Isolation Safety 000219
Transport Estimates of Hypothetical Radioactive Waste 000221
Transport Model for Long Term Release of Low-Level 000311
Transport Model: Theory and Numerical Implementation (000215
Transport Properties of Nuclear Waste in Geologic Media* 000298
Transport Theory to the Leaching of Radionuclides from 000258
Transport* Waste Isolation in Geologic Formations. 000237
Transport, Handling, and Interim Storage of 000187
Transportation of Liquid Metal Fast Breeder Reactor 000096
Transportation of Radioactive Waste and Spent Fuel* 000566
Transportation of Radioactive Waste-Materials to the Sea 000448
Transportation of Spent Fuel by Barge and By Rail: A 000565
Transportation Fires* 000110
Transportation Quarterly Progress Report, April Through 000333
Transportation Quarterly Progress Report, October 000226
Transportation System* @National Waste Terminal 000564
Transportation Systems Analysis and Program Plan* 000187
Transportation, and Disposal of Non-Fuel Bearing 000651
Transporting Solid Radioactive Wastes* the Organization 000567
Transuranic Elements in Soils, Plants and Animals* 000310
Transuranic Storage Area in Southeastern Idaho* 000181
Transuranic Waste at the Idaho National Engineering 000184
Transuranic Waste at the Idaho National Engineering 000187
Transuranic Waste at the Savannah River Plant, Aiken, 000206
Transuranic Waste from the Nuclear Fuel Cycle: January- 000570
Transuranic Waste from the Nuclear Fuel Cycle: July- 000571
Transuranic Waste Incinerator* 000650
Transuranic Waste Management on the Workers and on the 000209
Transuranic Waste Management* 000307
Transuranic Waste Management* @Identification of 000475
Transuranic Waste Storage Under Conditions Expected in 000087
Transuranic Waste* 000308
Transuranic Waste*Alternatives for Long-Term Management 000340
Transuranic Wastes Under Conditions Expected in the 000007
Transuranic Wastes Under Conditions Expected in the 000008
Transuranic Wastes Under Conditions Expected in the 000009
Transuranic Wastes* 000228
Transuranic-Contaminated Hydrogenous Waste* 000091
Transuranics in Environmental Water* 000600
Transuranics in High-Level Waste* 000700
Transuranium Nuclides on Container Surfaces* 000133
Treating Radioactive Wastes from Reprocessing 000625
Treatments Costs and the Environmental Impact of Waste 000501
Trench at Western New York Nuclear Service Center, West 000173
Trenches* @Concerning the 000357
Triaxial Compression up to 200 C* 000086
Tritiated Waste Forms* 000100
Tritium Monitor for Nuclear Facility Off-Gas Streams* 000172
Trivalent Cerium and Yttrium Cations* 000002
Trojan Nuclear Plant's Spent Resin Transfer System* 000658
TRU Waste from the Nuclear Fuel Cycle: October-December 000628
TRU Waste Cyclone Drum Incinerator and Treatment 000622
Tuff Units from the J-13 Drill Site, Jackass Flats, 000410
Tunnels and Test Rooms at Stripa* 000350
Twenty-Five Years Hence ... And Beyond* 000372
U.S. Atomic Energy Commission, Washington* and Ground- 000189
U.S. Department of Energy Defense Waste, July-December 000389
U.S. Geological Survey* 000576
U.S. Nuclear Regulatory Commission* 000512
U.S. Program for the Immobilization of High-Level 000586
U.S.A.* Search for Carbonates Suitable for Developing 000371
Ultimate Disposal of High-Level Waste* of Salt Domes, 000407
Ultimate Disposal* @ 000409
Ultimate Storage of High-Level Waste Solids and Liquids 000359
Ultimate Storage of High-Level Waste Solids and Liquids 000360
Ultrasonic Shear Waves* 000481
Ulva pertusa)* @Influences of Marine Sediment 000285
Underground Disposal of Canada's Nuclear Waste* 000470
Underground Emplacement of Waste* in Parts of Alabama, 000430
Underground In Situ Melting* @Economic and 000488
Underground Limestone Mines in the Ohio Region* 000370
Underground Mining Techniques for Removal of 000630
Underground Radioactive Waste Disposal in Argillaceous 000219
Underground Repositories of Radioactive Wastes* 000018
Underground Repositories of Radioactive Wastes* 000380
Underground Salt Mines* Salt Vault: A Demonstration 000316
Underground Storage of Bitumen Blocks Containing 000581
Underground Storage of Bitumen* 000202
Underground Storage of Liquid Radioactive Wastes in the 000668
Underground Storage of Radioactive Wastes*Salt Deposits 000817
Underground Storage Condition for Radioactive Wastes* 000005
Underground Storage Facilities* @Design Aspects 000561
Underground Storage Space in the Midcontinent, U.S.A.* 000371

PERMUTED TITLE INDEX

*Mining Methods Used in the
Plant Area* *
*Heat Pipe Cooling System for
Oklo and Other Examples* *Thermodynamic Considerations
*Characterization of Actinide-Bearing Sediments
Needs* *Science
*Autonomous Aspiration
*Radioactive Waste Management in the
Plutonium Concentrations in Deer from the Southeastern
*Waste Isolation in Geologic Formations in the
Bedies from the Appalachian Mountain Region of Eastern
Chemical Relationships in Salt Depositories* *
*Ion Exchange Research by the
*Projections of Spent Fuel to be Discharged by the
Geologic Disposal of Commercial Radioactive Wastes**The
Disposal Feasibility Study* *
241, Neptunium 237, and Technetium 99 Sorption on Two
*High-Level Nuclear Waste Management in the
*Petrology of Tuff
*Mineral Exchange at the
Contamination of Soils: Behavior of Radioactive Ions in
(IAEA) Definition of High-Level Radioactive Waste
*Let's Go Slowly on a Nuclear Power Program
of Technetium 99 and Neptunium 237 in Contact with
Evaluation of New Macroporous Resins for the Removal of
- Fabrication of Light-Water Reactor Fuel from Enriched
Wastes from the Uranium-Plutonium and Thorium-
the Nuclear Fuel Cycle - Conversion of Yellow Cake to
Cycle* *Migration of
of High-Temperature Gas-Cooled Reactor Fuel Containing
*Migration of Uranium Process Wastes from the
Contaminated Spent Fuel Reprocessing Wastes from the
Technetium, Cesium, Barium, Cerium, Europium,
Underground Storage of Liquid Radioactive Wastes in the
Emplacement Site, Salt Valley Anticline, Grand County,
Disposal of Radioactive Wastes*An Investigation of the
Geology of a Potential Waste Emplacement Site, Salt
Studies* *Tennessee
to *Geologic Investigation of the Virgin River
Trench at Western New York Nuclear Service Center, West
Fuel Services, Incorporated, Reprocessing Plant at West
*An Analysis of the Measured
*Compressible Fluid Flow Through Rocks of
Radionuclides in Hanford Subsoils* *
Solidified Waste in Underground Salt *Project Salt
*Let's Go Slowly on a Nuclear Power Program Until We'
Numerical Implementation (Discrete-Parcel-Random-Walk
Aquifer. 1. *Investigation of Salt Transport in
Aquifer. 2. *Investigation of Salt Transport in
Aquifer. 3. *Investigation of Salt Transport in
*Processing of High-Level Atomic Wastes with the
Southeastern Nevada to *Geologic Investigation of the
River Plant Waste* *
Composition Ranges and Properties* *
*
Experience with a Joule Heated Ceramic Melter for
of Radioactivity Release Due to Leaching of
*Handling of Spent Nuclear Fuel and Final Storage of
*Above-Ground Storage Facilities for
*Microstructures and Leachability of
Products* *Clay-Phosphate Ceramics and
Simulated High-Level Radioactive Waste* *
Spectrometry* *Investigations of the
*Confinement of
Mechanically Agitated Thin-Film Evaporators* *
Transuranic Waste from the Nuclear Fuel Cycle: January-*
Transuranic Waste from the Nuclear Fuel Cycle: July- *
from the Nuclear Fuel Cycle: October-December 1977* *
Disposal of Non-Fuel Bearing Hardware *Experience in
(GEIS): Radioactive Waste in Geologic Formations.
Storage Program, Configuration Management Plan.
Temperature Fields for the Stripa Heater Project,
and *Radionuclide Interaction with Soil and Rock Media.
: Radioactive Waste Isolation in Geologic Formations.
: Radioactive Waste Isolation in Geologic Formations.
: Radioactive Waste Isolation in Geologic Formations.
Radioactive Waste Isolation in Geologic Formations.
Radioactive Waste Isolation in Geologic Formations.
Assessment Method for Radioactive Waste Management.
Cycle with Emphasis on High-Level Waste Management.
: Radioactive Waste Isolation in Geologic Formations.
: Radioactive Waste Isolation in Geologic Formations.
Radioactive Waste Isolation in Geologic Formations.
Boreholes with Compacted Natural Earthen Materials,
: Radioactive Waste Isolation in Geologic Formations.
: Radioactive Waste Isolation in Geologic Formations.
*Information Base for Waste Repository Design,
: Radioactive Waste Isolation in Geologic Formations.

Underground Tunnels and Test Rooms at Stripa* 000350
Underground Waste Disposal Studies, Chemical Processing 000192
Underground, Radioactive Waste Storage Tanks* 000107
Underlying the Migration of Radionuclides in Geomedia: 000231
Underlying Liquid Waste Disposal Facilities at Hanford* 000280
Underlying Radioactive Waste Management: Status and 000662
Unit for Sampling Radioactive Aerosols from Air* 000182
United Kingdom* 000613
United States and in Deer from an Integrated Nuclear 000271
United States of America* 000528
United States with Regard to Siting of Radioactive 000888
United States Geological Survey Studies of Physical- 000079
United States Geological Survey* 000068
United States Nuclear Power Industry* 000690
United States Program for the Safety Assessment of 000219
United States Seabed High-Level Radioactive Waste 000308
United States Subsoils from Differing Weathering 000293
United States: A Time for Decisions* 000513
Units from the J-13 Drill Site, Jackass Flats, Nevada* 000810
University of North Carolina* 000677
Unsaturated Soils after Water Disposal into a Limited 000292
Unsuitable for Dumping at Sea* Atomic Energy Agency 00018*
Until We've Solved Waste Problems* 000527
Unweathered Igneous Rocks* *Chemistry 000228
Uranium and Plutonium from Waste Streams* * 000623
Uranium Dioxide* "As Low As Practicable" Guides 000652
Uranium Fuel Cycles* Spent Fuel Reprocessing 000035
Uranium Hexafluoride. Part 1. The Fluorination- 000521
Uranium Process Wastes from the Uranium 233-Thorium 232 000251
Uranium 233 and Thorium* Fuel Cycle - Reprocessing 000501
Uranium 233-Thorium 232 Cycle* 000251
Uranium-Plutonium and Thorium-Uranium Fuel Cycles* 000035
Uranium, Plutonium, and Americium*Studies of Strontium, 000028
USSR* *The Scientific Basis for, and Experience With, 000468
Utah* *Subsurface Geology of a Potential Waste 000411
Utility of Gulf Coast Salt Domes for the Storage or 000828
Valley Anticline, Grand County, Utah* *Subsurface 000411
Valley Authority Away-From-Reactor Spent Fuel Storage 000320
Valley Salt Deposits, Clark County, Southeastern Nevada 000439
Valley, New York* Analysis of Soil Samples from Test 000173
Valley, New York* Related to the Closing of the Nuclear 000169
Values for the State of Stress in the Earth's Crust* 000818
Variable Permeability* 000089
Variables Affecting Sorption and Transport of 000358
Vault: A Demonstration of the Disposal of High Activity 000316
ve Solved Waste Problems* 000527
Version)* Mass Transport Model: Theory and 000215
Vertical Boreholes and Brine Invasion into Fresh Water 000273
Vertical Boreholes and Brine Invasion into Fresh Water 000274
Vertical Boreholes and Brine Invasion into Fresh Water 000275
View to Ultimate Disposal* 000609
Virgin River Valley Salt Deposits, Clark County, 000439
Viscosity of Glasses Containing Simulated Savannah 000657
Vitrification of High Sodium-Aluminum Wastes: 000104
Vitrification of High-Level ICPP Calcined Wastes* 000602
Vitrification of High-Level Liquid Waste* Operation 000686
Vitrified High Level Waste* *Calculations 000591
Vitrified High-Level Reprocessing Waste. 4. Safety 000123
Vitrified Highly Active Wastes* 000573
Vitrified Radioactive Wastes* 000659
Vitromets: Alternatives to Monolithic High Level Waste 000689
Volatility of a Zinc Borosilicate Glass Containing 000608
Volatility of Cesium from Aluminosilicates by Mass 000138
Volatilized Ruthenium Oxides* 000637
Volume Reduction of Liquid Radioactive Wastes Using 000592
Volume Reduction System for Solid and Liquid 000570
Volume Reduction System for Solid and Liquid 000571
Volume Reduction System for Solid and Liquid TRU Waste 000638
Volume Reduction, Packaging, Transportation, and 000651
Volume 1. Executive Summary* Impact Statement 000552
Volume 1. Management Summary* *National Waste Terminal 000499
Volume 1* *Theoretical 000378
Volume 1: Processes Influencing Radionuclide Mobility 000217
Volume 10. Repository Preconceptual Design Studies: 000587
Volume 12. Repository Preconceptual Design Studies: 000588
Volume 18. Repository Preconceptual Design Studies: 000548
Volume 16. Repository Preconceptual Design Studies: 000550
Volume 19 - Thermal Analyses* Impact Statement: 000075
Volume 2. Implementation for Terminal Storage in 000279
Volume 2* Analysis of the Back End of the Nuclear Fuel 000321
Volume 20. Thermomechanical Stress Analysis and 000028
Volume 21. Ground Water Movement and Nuclide Transport* 000237
Volume 23 - Environmental Effluent Analyses* Statement: 000296
Volume 3* *Feasibility of Sealing 000066
Volume 4. Baseline Rock Properties: Salt* (GEIS) 000388
Volume 6. Baseline Rock Properties - Shale* (GEIS) 000022
Volume 6. Excavation Technology* 000543
Volume 7. Baseline Rock Properties - Basalt* (GEIS) 000023

PERMUTED TITLE INDEX

: Radioactive Waste Isolation in Geologic Formations.
Boreholes with Compacted Natural Earthen Material,
and Numerical Implementation (Discrete-Parcel-Random-
@Preliminary Concept of Waste Retrieval System-Tank
Level Radioactive Waste Hanford Observations, Richland,
Reservation of the U.S. Atomic Energy Commission,
Characterization of an Improved High Level Radioactive
Mining Techniques for Removal of Radioactive
@Radioactive
@Logistics Models for the Transportation of Radioactive
@Interactions between Nuclear
@Retrieval of Buried Transuranic
and Interim Storage of Intermediate-Level Transuranic
for Long-Term Management of Defense Transuranic
@Sampling of Stored High-Level Radioactive Calcined
@Handling of Radioactive
@Fixation of Radioactive
Rock* @Disposal of Nuclear
@The Fixation of Radioactive
Reduction System for Solid and Liquid Transuranic
Reduction System for Solid and Liquid Transuranic
* @Volume Reduction System for Solid and Liquid TRU
Chemical Properties* @Sources of Radioactive
@The Disposal of High-Level Nuclear
@A Concept for Long-Term Isolation of Solid Radioactive
Environmental Impact Statement (GEIS): Radioactive
Protection Aspects of Disposal of High-Level
@Transport Properties of Nuclear
Phosphate and Borosilicate @Fixation of Radioactive
of Actinide Critical Masses from Stored Nuclear
@Concerning the Possibility of Disposing of Radioactive
Protection Aspects of Disposal of High-Level
Radionuclide Repository in @Putting Radioactive
@The Disposal of Radioactive
Concerning the Evolution in the Ground of Radioactive
@Long Term Geochemical Interactions of High Active
@An Evaluation of the Storage of Radioactive
@Interaction Between Processing Method and Radioactive
Environmental Assessment Related to Hanford Radioactive
@Stabilization of Simulated Commercial
Isolation* @High-Level
@High-Level
@History of Prototype High Level
@Nuclear
Test of a Glass-Ceramic Candidate for a Nuclear
Assay System for the Idaho Chemical Processing Plant
@Improved Heat Transfer from Radioactive
Physical and Chemical Testing* @Low-Level Solidified
@Cermets for High Level
July-Septemb.. 1978* @TRU
Reprocessing Plant* @Radioactive
Facilities* @Radioactive
@Ocean
National Laboratory* @
National Laboratory* @
@Economic and Environmental Evaluation of Nuclear
@Possibilities for Subsurface
@The Research and Development Program on
* @Radioactive
Some @An Appraisal of Underground Radioactive
@Environmental Aspects of High-Level Radioactive
Development, @A Model for Calculating Effects of Liquid
@Long-Term Risk Assessment of Radioactive
@The Rock-Melt Approach to Nuclear
@Radioactive
@Geology of Subsurface
Abstracts* @Radioactive
@Radiological Problems of Liquid
@A Review of Radioactive
Leakage Studies on Natural and Synthetic Rock Salt for
@Nuclear Shipping and
of Actinide-Bearing Sediments Underlying Liquid
@United States Seabed High-Level Radioactive
An Interim Report* @Hydrology of
@Oceanographic Research on
@Modelling Studies Used to Evaluate
@Dissolution of Kansas Evaporites - The Radioactive
@
@Estimate Magnitude of
Column Experiment* @Radioactive
at Commercially Operated Low-Level Radioactive
at Commercially Operated Low-Level Radioactive
Water Chemistry at Commercially Low-Level Radioactive
@Underground
Nuclear Generating Plant Liquid and Solid Radioactive
@High-Level and Long-Lived Radioactive
@Salt Motion Following Nuclear
@A Bibliography on Ocean

Volume 8. Repository Preconceptual Design Studies: Salt* 000545
Volumes 1 and 2* @Feasibility of Sealing 000052
Walk Version)* Mass Transport Model: Theory 000215
Wall Access Using Underground Mining Techniques for 000630
Washington* for Long-Term Management of Defense High- 000387
Washington* Ground-Water Characteristics of the Hanford 000189
Waste (HAW) Borosilicate Glass* @Preparation and 000605
Waste (Sludge, Saltcake and Liquid)* Using Underground 000630
Waste "Graveyards"* 000445
Waste and Spent Fuel* 000566
Waste and Surrounding Rock* 000282
Waste at the Idaho National Engineering Laboratory* 000188
Waste at the Idaho National Engineering Laboratory* 000187
Waste at the Savannah River Plant, Aiken, South Carolina 000206
Waste at Idaho Chemical Processing Plant* 000158
Waste at Soviet Nuclear Power Stations* 000628
Waste by Hydrothermal Reactions with Clays* 000575
Waste by In-Situ Incorporation in Deep Molten Silicate 000632
Waste for Posterity* 000805
Waste from the Nuclear Fuel Cycle: January-March 1978* 000570
Waste from the Nuclear Fuel Cycle: July-September 1978* 000571
Waste from the Nuclear Fuel Cycle: October-December 1977 000628
Waste from Light-Water Reactors and Physical and 000691
Waste in the Oceans* 000877
Waste in Fine-Grained Deposits* 000376
Waste in Geologic Formations. Volume 1. Executive 000552
Waste in Geologic Formations* of the Radiological 000268
Waste in Geologic Media* 000298
Waste in Glass. Part 2. The Experimental Evaluation of 000109
Waste in Natural Rock Repositories* on Accumulation 000015
Waste in Ocean Trenches* 000357
Waste on the Ocean Floor"Assessment of the Radiological 000261
Waste on Ice - A Proposal for an International 000892
Waste or Land* 000871
Waste over a Long Period* Oklo Natural Nuclear Reactor 000287
Waste with Crystalline Rock Repository Media* 000157
Waste within Silicic Pyroclastic Rocks* 000077
Waste Burial Site* 000463
Waste Burial* 000288
Waste Calcine* 000660
Waste Canister Corrosion Studies Pertinent to Geologic 000107
Waste Canister Emplacement and Retrieval Concepts Study* 000561
Waste Canister SS-9 While in Air and Water Storage* 000101
Waste Canister Thermally Induced Motion* 000025
Waste Canister* @Preliminary Corrosion 000151
Waste Canisters and Spent Fuels Preliminary Description 000128
Waste Canisters* 000119
Waste Characterization: The Scientific Basis for 000003
Waste Containment* 000569
Waste Cyclone Drum Incinerator and Treatment System: 000622
Waste Discharges to the Environment from a Nuclear Fuel 000622
Waste Discharges to the Environment from Nuclear Power 000280
Waste Disposal (A Bibliography with Abstracts)* 000367
Waste Disposal by Shale Fracturing at Oak Ridge 000213
Waste Disposal by Shale Fracturing at Oak Ridge 000214
Waste Disposal by Underground In Situ Melting* 000468
Waste Disposal in a Structural Syncline in Pennsylvania* 000405
Waste Disposal in the Federal Republic of Germany* 000161
Waste Disposal in Antarctica: Some Glaciological Aspects 000881
Waste Disposal in Argillaceous and Crystalline Rocks: 000218
Waste Disposal in Deep Geologic Formations* 000528
Waste Disposal in Deep Saline Aquifers. Part 1 -- 000267
Waste Disposal in Geologic Formations* 000256
Waste Disposal in Geologic Media* 000017
Waste Disposal in Geological Formations* 000802
Waste Disposal in Michigan Basin* 000368
Waste Disposal in Salt Deposits - A Bibliography with 000836
Waste Disposal into the Sea* 000281
Waste Disposal to the Ground at Hanford* 000191
Waste Disposal Applications* @Radiation 000081
Waste Disposal Cost Estimates* 000507
Waste Disposal Facilities at Hanford* @Characterization 000290
Waste Disposal Feasibility Study* 000206
Waste Disposal National Reactor Testing Station, Idaho, 000819
Waste Disposal Off the Coast of Tokai-Mura* 000815
Waste Disposal Options* 000260
Waste Disposal Problem* 000459
Waste Disposal Problems* 000331
Waste Disposal Problems* 000698
Waste Disposal Report on Seepage Pit Liquid Waste-Shale 000179
Waste Disposal Sites* Land Burial Water Chemistry 000166
Waste Disposal Sites* Land Burial Water Chemistry 000168
Waste Disposal Sites* of Tritope Migration-Land Burial 000167
Waste Disposal Studies, Chemical Processing Plant Area* 000192
Waste Disposal System* @Operation of the Monticello 000595
Waste Disposal* 000351
Waste Disposal* 000386
Waste Disposal* 000865

PERMUTED TITLE INDEX

Review of Rock Mechanics Workshop on Radioactive Waste Disposal* @Summary 000373
Mineral Alteration: An Implication for Radioactive Waste Disposal* @Botanict 000030
an Effective Aluminosilicate Absorbent for Radioactive Waste Disposal* @Development of 000606
Geological Occurrences and in Deep-Burial Radioactive Waste Disposal* @Pollucite and Its Alteration on 000105
@Are Plutons the Answer to Nuclear Waste Disposal?* 000318
@Nuclear Waste Disposal: The Geological Aspects* 000375
@Release of Radionuclides from High-Level Waste During Transportation Fires* 000110
Waste Treatments Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - 000501
Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Conversion 000521
Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in 000652
County, Utah* @Subsurface Geology of a Potential Waste Emplacement Site, Salt Valley Anticline, Grand 000411
@Liquid Waste Evaporation and Acid Recovery* 000666
Plant* @Description of Solid Waste Expected from a Large Commercial Reprocessing 000702
Temperature and Pressure) Concretes - @Radioactive Waste Fixation in PUREX (Formed Under Elevated 000060
Plant Zirconia Wastes* @Development of a Pelleted Waste Form for High-Level Idaho Chemical Processing 000628
@Hydrothermal Reactivity of Simulated Nuclear Waste Forms and Water-Catalyzed Waste-Rock Interactions* 000054
Design Studies: Battelle Pacific Northwest Laboratory Waste Forms in Salt*Volume 16. Repository Preconceptual 000550
@Radiolytic Gas Production from Tritiated Waste Forms* 000100
@Natural Glasses: Analogues for Radioactive Waste Forms* 000111
@Raman Spectra of Cesium Aluminosilicate Waste Forms* 000113
@Long-Term Elevated Temperature Leaching of Solid Waste Forms* 000124
@Leach Rate Characterization of Solid Radioactive Waste Forms* 000246
@Characterization of Glass and Glass Ceramic Nuclear Waste Forms* 000635
@Development of Multibarrier Nuclear Waste Forms* 000665
Procedure for Determining Leachabilities of Radioactive Waste Forms* 000044
Prediction* @Chemistry of Nuclear Waste Glass Reactions: Problems and Potential of 000093
@Leaching of Fully Radioactive High-Level Waste Glass* 000358
@High-Level Waste Glass* 000643
Devitrification Behavior in a Zinc Borosilicate Nuclear Waste Glass* 000679
Characteristics of Actinides from Simulated Reactor Waste Glass* @Leaching 000153
of Leached Surface Layers on Simulated High-Level Waste Glasses by Sputter-Induced Optical Emission* 000118
@Annual Report on the Characterization of High-Level Waste Glasses* 000661
from Alpha-Decay and its Significance for Radioactive Waste Glasses* @Helium Formation 000131
Facilities at the New Hungarian Radioactive Waste Handling and Disposal Station* @Technological 000597
long-Term Management of Defense High-Level Radioactive Waste Hanford Reservations, Richland, Washington* for 000347
long-Term Management of Defense High-Level Radioactive Waste Idaho Chemical Processing Plant, Idaho Falls, 000386
Transport Estimates of Hypothetical Radioactive Waste Incidents* @Hanford Groundwater 000221
Production, and Reprocessing Division @Status Report, Waste Incineration and Fixation for Waste Management, 000490
Facilities (April-September 1977)* Waste Incineration and Immobilization for Nuclear 000616
@Operation of Low-Level Radioactive Waste Incinerator* 000584
of a Criticality Monitoring Technique for a Transuranic Waste Incinerator* @Selection 000650
States of America* Waste Isolation in Geologic Formations in the United 000528
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Baseline Rock 000021
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Facility 000586
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Stratigraphics 000385
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 10. 000507
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 12. 000588
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 14. 000589
for Generic Environmental Impact Statement: Radioactive Waste Isolation in Geologic Formations. Volume 16. 000550
for Generic Environmental Impact Statement: Radioactive Waste Isolation in Geologic Formations. Volume 19 - 000075
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 20. 000028
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 21. 000237
for Generic Environmental Impact Statement: Radioactive Waste Isolation in Geologic Formations. Volume 23 - 000296
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 4. 000388
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 6. 000022
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 7. 000023
Environmental Impact Statement (GEIS): Radioactive Waste Isolation in Geologic Formations. Volume 8. 000585
Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations* Impact 000457
@Corrosion Considerations for Nuclear Waste Isolation Canisters* 000102
@Description of Waste Isolation Facilities* 000553
in Salt Domes and How It Affects a Radioactive Waste Isolation Facility Design* to Dissolution 000263
@Effluents from the Waste Isolation Facility* 000558
WIPP) Site, Southeastern New Mexico. Special Studies of Waste Isolation Pilot Plant (WIPP) Rocks* Pilot Plant (000073
of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico* 000281
New Mexico. @Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern 000073
New Mexico. @Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern 000078
New Mexico* @Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern 000338
New Mexico. @Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern 000453
New Mexico. Site @Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern 000454
New Mexico. @Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern 000455
New Mexico. @Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern 000456
on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP). December 15, 1978- 000007
on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP). October 1-December 000008
on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), March 15-June 15, 000009
Waste Storage Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), October 1-December 000047
@New Mexico Waste Isolation Pilot Plant - A Status Report* 000482
to Estimate Migration Potential at the Proposed Waste Isolation Pilot Plant in New Mexico* of Nuclides 000300
@Preliminary Safety Assessment of the Waste Isolation Pilot Plant Facility* 000494
Program for High-Level Waste* Waste Isolation Pilot Plant In Situ Experimental 000137
@Forced-Air Cooling of a Waste Isolation Pilot Plant Mine Drift* 000532
@Environmental Monitoring at the Waste Isolation Pilot Plant Site, Calendar Year 1976* 000363
@Revised Concept for the Waste Isolation Pilot Plant* 000537
Salado Salt, New Mexico, to Problems of Siting the Waste Isolation Pilot Plant*Fluid Inclusions in Perplas 000072
Degradation of RAM-Wastes for the Sandia Laboratories Waste Isolation Pilot Project (WIPP), April 1-June 30, 000160
@Basalt Waste Isolation Program Annual Report* 000451
@Basalt Waste Isolation Program Monthly Report, April 1978* 000390
@Basalt Waste Isolation Program Monthly Report, February 1978* 000387
@Basalt Waste Isolation Program Monthly Report, January 1978* 000388

PERMUTED TITLE INDEX

	@Result	Waste Isolation Program Monthly Report, March 1978* 000389
@Material Control and Accountability Procedures for a		Waste Isolation Repository* 000120
Control and Accountability Considerations for a		Waste Isolation Repository* @An Overview of Material 000496
of Lawrence Berkeley Laboratory Contract No. 4590UAK.		Waste Isolation Safety Assessment Program - Collection 000219
Description of Source Term	@Overview of	Waste Isolation Safety Assessment Program and 000315
Admixtures in Preparing Dose Counts for Radioactive		Waste Isolation* @Role of 000662
Measurements on Constitution Materials for Nuclear		Waste Isolation* @Permeability 000156
	@High-Level	Waste Leakage from the 241-T-106 Tank at Hanford* 000198
Administration Technological Program for Radioactive		Waste Management and Reprocessing* and Development 000325
Report, October Through December 1975*	@Nuclear	Waste Management and Transportation Quarterly Program 000226
Report, April Through June 1976*	@Nuclear	Waste Management and Transportation Quarterly Program 000333
	@Radioactive	Waste Management in the United Kingdom* 000613
Decisions*	@High-Level Nuclear	Waste Management in the United States: A Time for 000513
	@Radioactive	Waste Management in Japan* 000509
	@	Waste Management of Actinide Contaminated Soil* 000640
	@Effects of Current Transuranic	Waste Management on the Workers and on the Public* 000209
Parts 1 and 2: Program Report for Period April 1 to	@	Waste Management Analysis for the Nuclear Fuel Cycle- 000647
Selection of Significant Elements and Radionuclides for		Waste Management Assessment* @ 000703
at the Los Alamos Scientific Laboratory*	@	Waste Management Capabilities for Alpha Bearing Wastes 000419
	@The Radioactive	Waste Management Complex and Its Operations* 000208
Engineering @Safety Review Document for the Radioactive		Waste Management Complex at the Idaho National 000176
	@Systems Analysis Study for	Waste Management Criteria* 000505
	@Radioactive	Waste Management Development in Europe* 000688
	@Guide to Radioactive	Waste Management Literature* 000323
	@Development of Advanced	Waste Management Plans* 000725
Testing Station*	@High-Level Radioactive	Waste Management Program at the National Reactor 000671
	@The High-Level Radioactive	Waste Management Program at Nuclear Fuel Services* 000589
Los Alamos Scientific Laboratory*	@	Waste Management Programs and Capabilities within the 000171
September 1976*	@	Waste Management Programs Quarterly Report, July- 000672
Oak Ridge National Laboratory*	@High-Level	Waste Management Research and Development Program at 000579
Battelle Pacific*	@High-Level Radioactive	Waste Management Research and Development Program at 000644
	@Alternative	Waste Management Systems* 000318
Application of a Risk Assessment Method for Radioactive		Waste Management. Volume 2. Implementation for Terminal 000179
of the Nuclear Fuel Cycle with Emphasis on High-Level		Waste Management. Volume 2*@An Analysis of the Back End 000321
	@Nuclear	Waste Management* 000190
	@Scientific Basis for Nuclear	Waste Management* 000330
@Nuclear Regulatory Commission Program for High Level		Waste Management* 000508
	@Nuclear	Waste Management* 000528
@Relevance of Natural Retention Experience to Nuclear		Waste Management* 000588
	@High-Level Radioactive	Waste Management* 000582
@Nuclear Waste Management--Review of Commercial Nuclear		Waste Management* 000610
Effects from Current Operations of Transuranic		Waste Management* @Environmental 000307
Estimating Accidental Radioactive Releases in Nuclear		Waste Management* @Methodology for 000278
Process and Storage Materials Suitable for Krypton 85		Waste Management* @The Development of 000181
of Environmental Risks Associated with Radioactive		Waste Management* @An Assessment Methodology 000259
for Idaho National Engineering Laboratory Transuranic		Waste Management* @Identification of Alternatives 000475
Agency Program for Reprocessing and High Level		Waste Management* of the Environmental Protection 000328
Analysis of Separation Methods Applied to Radioactive		Waste Management* @Technology Assessment and Impact 000601
Management*	@Nuclear	Waste Management--Review of Commercial Nuclear Waste 000610
@Savannah River Laboratory Quarterly Report,		Waste Management, January-March 1975* 000668
@Savannah River Laboratory Quarterly Report,		Waste Management, July-September, 1975* 000199
of @Status Report, Waste Incineration and Fixation for		Waste Management, Production, and Reprocessing Division 000490
Generic Environmental Impact Statement on Commercial		Waste Management: Radioactive Waste Isolation in 000557
@Science Underlying Radioactive		Waste Management: Status and Needs* 000663
	@Radioactive	Waste Management: The Need for Multiple Barriers* 000150
their Suitability for Possible Storage of Radioactive		Waste Material as of September, 1977.* to Investigate 000439
Their Suitability for Possible Storage of Radioactive		Waste Material* Texas Salt Dome Basin to Investigate 000438
@Radiolytic Gas Generation in Plutonium Contaminated		Waste Materials* 00008[illegible]
November 1976-October 1977*	@Studies of Nuclear-	Waste Migration in Geologic Media: Annual Report, 00029*
Disposal*	@Synthesis of Nuclear	Waste Monazites, Ideal Actinide Hosts for Geologic 000680
	@Radioactive	Waste Package Development at the Rocky Flats Plant* 000136
@Development of National Waste Terminal Storage Program		Waste Packaging Criteria* 000127
@Gold and Selective Storage Solves Nuclear		Waste Problem* 000159
of the Activity Levels in Fusion Reactor Blankets.		Waste Problem* @Evaluation 0006*9
So Slowly on a Nuclear Power Program Until We've Solved		Waste Problems* @Let's 000527
	@Radioactive	Waste Processing* 000476
	@Radioactive	Waste Processing* 000682
and Vitrocets: Alternatives to Monolithic High Level		Waste Products* @Clay-Phosphate Ceramics 000689
	@Interactions of Simulated	Waste Radionuclides and Rocks* 000039
Progress*	@Interaction of	Waste Radionuclides with Geomedia: Program Approach and 000071
@Microstructural Interactions of Geologic Media with		Waste Radionuclides* 000038
@Radiation Doses to Students Inhabiting a Radioactive		Waste Receiving Area* 000283
1988 to 1967*	@A Review of Studies on Soil-	Waste Relationships on the Hanford Reservation from 000197
@Public Health Implications of Radioactive		Waste Releases* 000339
and Thermal Design Considerations for Radioactive		Waste Repositories in Hard Rock. Parts 1 and 2* 000019
@Geological Criteria for Radioactive		Waste Repositories* 000368
@State/Federal Licensing Interactions for Radioactive		Waste Repositories* 000508
and Release of Radiation Energy in Salt in Radioactive		Waste Repositories* @Storage 000082
United States with Regard to Siting of Radioactive		Waste Repositories* Mountain Region of Eastern 000088
the Geological Stability of a Hypothetical Radioactive		Waste Repository in a Bedded Salt Formation*Analysis of 000562.
	@A	Waste Repository in a Geologic Formation* 000529
the Deformation Behavior of a High-Level Radioactive		Waste Repository in Bedded Salt* Model for Simulating 000620
*	@Information Base for	Waste Repository Design, Volume 6. Excavation Technology 000583
@Deep Exploratory Test Holes for		Waste Repository Investigations* 000026
for the Suitability of Anhydrite as a Nuclear		Waste Repository Media* Brine and Their Implications 000016
of Kriging Techniques for High-Level Radioactive		Waste Repository Site Characterization* @An Evaluation 000026
Cement Technology for Plugging Boreholes in Radioactive		Waste Repository Sites: Progress Report for the Period 000059
Enhanced "Release Scenario" Analysis for a Nuclear		Waste Repository* @Computer 000560

PERMUTED TITLE INDEX

Conditions Near the Site of a Potential Nuclear	Waste Repository, Eddy and Lea Counties, Mexico*	000431
*Early	Waste Retrieval Interim Report*	000186
Underground Mining Techniques *Preliminary Concept of	Waste Retrieval System-Tank Wall Access Using	000630
Idaho Chemical Processing Plant High Level Radioactive	Waste Service* Evaluation of Stainless Steels in	000117
*Development of High-Level	Waste Shipping Cask Models*	000112
*Interaction of a Glass-Based Nuclear	Waste Simulant and Natural Rock*	000040
and Analyses of Savannah River Plant High-Level	Waste Sludges* *Sampling	000469
*Techniques for High-Level	Waste Solidification in Europe*	000611
99 and Selenium 79 in Nuclear Fuel Reprocessing and	Waste Solidification Facility Off-Gas* of Technetium	000598
Sandia Solidification Process - A Broad Range Aqueous	Waste Solidification Method* *The	000636
85 and Iodine 129 in Nuclear Fuel Reprocessing and	Waste Solidification Off-Gas* of Carbon 14, Krypton	000285
*Synthetic Rutile Microencapsulation: A Radioactive	Waste Solidification System Resulting in an Extremely	000597
*Current High-Level	Waste Solidification Technology*	000580
*Measurement and Control of Cement Set Times in	Waste Solidification*	000674
of a Titanate Based Ceramic for High Level Nuclear	Waste Solidification* *Characterization	030122
*Ultimate Storage of High-Level	Waste Solids and Liquids in Salt Formations*	000359
*Ultimate Storage of High-Level	Waste Solids and Liquids in Salt Formations*	030360
Transport Theory to the Leaching of Radionuclides from	Waste Solids* *Application of Mass	000258
Cesium and Strontium from *Processing of Radioactive	Waste Solution with Zeolite (Z). Leachabilities of	000619
*Removal of Plutonium from High-Level Caustic	Waste Solutions Using Bone Char - Pilot Study*	000578
Reference to the Subsurface Disposal of Radioactive	Waste Solutions* Summary of the Appalachian Basin with	000379
*Swedish-American Cooperative Program on Radioactive	Waste Storage in Mined Caverns in Crystalline Rock,	000087
of Allied-General Nuclear Services On-Site Solid	Waste Storage Concepts* *Description	000205
*Mathematical Model of	Waste Storage Operations*	000667
of the Geologic and Hydrologic Factors Related to the	Waste Storage Potential of Mesozoic Aquifers in the	000365
*Linearity Study of the Terminal	Waste Storage Program, Elkana Investigation*	000396
*Heat Pipe Cooling System for Underground, Radioactive	Waste Storage Tanks*	000107
*Investigation of Cooling Coil Corrosion in Radioactive-	Waste Storage Tanks*	000139
Isolation Pilot Plant (WIPP). *Studies of Transuranic	Waste Storage Under Conditions Expected in the Waste	000087
Resins for the Removal of Uranium and Plutonium from	Waste Streams* *Evaluation of New Macroporous	000623
Plutonium, and Ruthenium from Savannah River Plant	Waste Supernate* of Methods for Removing Strontium,	000588
Period October 1, 1976, to September 30, 1977**National	Waste Terminal Storage Program Progress Report for	000273
* *National	Waste Terminal Storage Program Progress Report June 1977	000312
*Development of National	Waste Terminal Storage Program Waste Packaging Criteria*	000127
Geologic Site Selection Factors For the National	Waste Terminal Storage Program* *Preliminary	000088
Achievable (ALARA) Studies Relative to The National	Waste Terminal Storage Program**As Low as is Reasonably	000403
Management Plan, Volume 1. Management Summary**National	Waste Terminal Storage Program, Configuration	000499
the Waste Transportation System* *National	Waste Terminal Storage Program: Potential Problems in	000568
Dissolution* *National	Waste Terminal Storage Review Meeting on Salt	000380
of Lawrence *Theoretical and Experimental Evaluation of	Waste Transport in Selected Rocks: 1977 Annual Report	000279
Terminal Storage Program: Potential Problems in the	Waste Transportation System* *National Waste	000568
*Radioactive	Waste Transportation Systems Analysis and Program Plan*	000187
Waste Effluents in the *Correlation of Radioactive	Waste Treatment Costs and the Environmental Impact of	000521
Waste Effluents in the *Correlation of Radioactive	Waste Treatment Costs and the Environmental Impact of	000652
*Correlation of Environmental Impact with Cost of	Waste Treatment*	000500
Waste Effluents in the *Correlation of Radioactive	Waste Treatments Costs and the Environmental Impact of	000501
Agency (IAEA) Definition of High-Level Radioactive	Waste Unsuitable for Dumping at Sea* Atomic Energy	000188
*Feasibility of Space Disposal of Radioactive Nuclear	Waste. 1. Executive Summary*	000434
*Feasibility of Space Disposal of Radioactive Nuclear	Waste. 2. Technical Summary*	000433
and Final Storage of Vitrified High-Level Reprocessing	Waste. 4. Safety Analysis* of Spent Nuclear Fuel	000123
*Shipments of Nuclear Fuel and	Waste...Are They Really Safe?*	000152
*Rock Stresses Associated with Burial of Nuclear	Waste*	000085
*An Improved Receptacle for Radioactive	Waste*	000142
*Incentives for Partitioning High-Level	Waste*	000233
*Underground Disposal of Canada's Nuclear	Waste*	000470
*Radioactive	Waste*	000516
*Management of Commercial High-Level Radioactive	Waste*	000551
*Secure Storage of Radioactive	Waste*	000587
*Transuranics in High-Level	Waste*	000700
for Solidification of Savannah River Plant High-Level	Waste* *Plan	000615
Actinide Removal on the Long-Term Hazard of High-Level	Waste* *Effect of	000585
of Surfactants on Bituminization of Radioactive	Waste* *Influence	000626
Challenges in the Immobilization of Hanford Defense	Waste* *Technical	000627
of Glasses Containing Simulated Savannah River Plant	Waste* *Viscosity	000657
Concrete as a Host for Savannah River Plant Radioactive	Waste* *Studies of	000673
of Glass Compositions for Immobilization of SRP	Waste* *Development	000656
Decontamination of Alkaline Solutions of Nuclear	Waste* *Ion Exchange	000406
Effects of Alternatives for Managing Transuranic	Waste* *Environmental	000300
of Environmental Protection Criteria for Radioactive	Waste* *Considerations	000526
Pilot Plant In Situ Experimental Program for High-Level	Waste* *Waste Isolation	000137
Scanning System for the Measurement of Plutonium in	Waste* *An Automatic Drum	000106
Gamma Sources in Light Water Reactor High-Level Nuclear	Waste* *Neutron and Gamma-	000590
to the Long Time Storage Problem for High-Level	Waste* *A Possible Solution	000628
to Work: Concentration of Aqueous Radioactive	Waste* *Setting Evaporators	000603
Compositions for Immobilization of Savannah River Plant	Waste* *Development of Glass	000658
as a Matrix for Solidification of Savannah River Plant	Waste* *Evaluation of Concrete	000667
for the Consolidation and Containment of Radioactive	Waste* *Hot Isostatic Pressing	000629
Disposal Site for the World's High-Level Radioactive	Waste* *Antarctica, A Potential	000491
Plan for Mined Geologic Disposal of Radioactive	Waste* *Earth Science Technical	000503
from Concrete Containing Savannah River Plant	Waste* *Radiolytic Gas Production	000090
Long-Term Management of Defense High-Level Radioactive	Waste* *Summary - Alternatives for	000470
Release Due to Leaching of Vitrified High Level	Waste* *Calculations of Radioactivity	000591
Attack of Transuranic-Contaminated Hydrogeneous	Waste* *Gas Generation from Radiolytic	000091
and Shaft Sealing for Geologic Isolation of Radioactive	Waste* *The States of Borehole Plugging	000536
*Glass Containing Simulated High-Level Radioactive	Waste* *Volatility of a Zinc Borosilicate	000680
on Borosilicate Glass Containing Savannah River Plant	Waste* *Effect of Internal Alpha Radiation	000099
Americium/Curium Separated from High-Level Reprocessing	Waste* *Final Purification and Concentration of	000409
Ceramic Melter for Vitrification of High-Level Liquid	Waste* Operation Experience with a Joule Heated	000686

PERMUTED TITLE INDEX

of Shipping Casks for Spent Fuel and High-Level	Waste*	
their Exploitation for Ultimate Disposal of High-Level	Waste*	
as Potential Host Rocks for Underground Emplacement of	Waste*	
and Pressure to Repositories for Radioactive	Waste*	
the Safety of the Geologic Isolation of Nuclear	Waste*	
from Nuclear Facilities and General Radioactive	Waste*	
Idaho National Engineering Laboratory Transuranic	Waste*	
of Glasses Containing Savannah River Plant	Waste*	
on the Sleeve* ©Thermoplastic/Plastic Analysis of	Waste-Container Sleeve: 1. Initial Estimates of Loading	
on Sleeve Loading* ©Thermoplastic/Plastic Analysis of	Waste-Container Sleeve: 3. Influence of Salt Strength	
Closure* ©Thermoplastic/Plastic Analysis of	Waste-Container Sleeve: 4. Air Gap Influence on Hole	
©Subsurface	Waste-Disposal Potential in Salina Basin of Kansas*	
©	Waste-Isolation Projects: Fiscal Year 1978*	
©International	Waste-Management Symposium*	
©Transportation of Radioactive	Waste-Materials to the Sun*	
©	Waste-Rock Interactions and Bedrock Reactions*	
©	Waste-Rock Interactions*	
of Simulated Nuclear Waste Forms and Water-Catalyzed	Waste-Rock Interactions* ©Hydrothermal Reactivity	
Radioactive Waste Disposal Report on Seepage Pit Liquid	Waste-Shale Column Experiment* ©	
Neozoic Aquifer System Below ©Geologic Evaluation of	Waste-Storage Potential in Selected Segments of the	
System for U.S. Department of Energy Defense	Waste, July-December 1977* ©Fluidized Bed Incineration	
©Radionuclide Migration from Low-Level	Waste: A Generic Overview*	
©Nuclear	Waste: Increasing Scale and Sociopolitical Impacts*	
©Nuclear	Waste: Twenty-Five Years Hence ... And Beyond*	
Characterization of Solidified Simulated Reactor	Wasteforms* ©Microstructural	
©Geologic Disposal of High-Level Radioactive	Wastes - Earth-Science Perspectives*	
Influence of Radiation on the System Liquid Radioactive	Wastes - Geologic Formation* ©	
Storage* ©Maximum Acceptable Temperatures of	Wastes and Containers During Retrievable Geologic	
©Recent Experiments on the Treatment of Medium Level	Wastes and Spent Solvent and on Fixation into Bitumen*	
©Waste Management Capabilities for Alpha Bearing	Wastes at the Los Alamos Scientific Laboratory*	
Experience from Plant ©Bituminization of Radioactive	Wastes at the Nuclear Research Center Karlsruhe -	
©System Design for Retrieval of Solidified High-Level	Wastes at Hanford*	
Agents used and Products Formed with Radioactive	Wastes at LWR Nuclear Power Plants* of Solidification	
Research Needed for Safe Disposal of Radioactive	Wastes at Sea* ©Oceanic	
©Subsurface Disposal of Liquid Industrial	Wastes by Deep-Well Injection*	
of Strontium and Cesium in Intermediate-Level Liquid	Wastes by Solidification in Cements* ©Insobilization	
Pilot ©Experimental Studies of the Degradation of BAP-	Wastes for the Sandia Laboratories Waste Isolation	
©Management of Highly Active Decladding Zircaloy Solid	Wastes from a Fuel Reprocessing Plant*	
©Migration of Uranium Process	Wastes from the Uranium 233-Thorium 232 Cycle*	
Level and Actinide-Contaminated Spent Fuel Reprocessing	Wastes from the Uranium-Plutonium and Thorium-Uranium	
Relative Areas for the Disposal in Bedded Salt of LWR	Wastes from Alternative Fuel Cycles* ©Estimates of	
©Long-Term Planning for Management of Aqueous	Wastes from Fuel Reprocessing Plants*	
©Management of Nontritium Radioactive	Wastes from Fusion Power Plants*	
Bitumen* ©Incorporation of Radioactive	Wastes from Nuclear Power Plants into Concrete and	
©Studies on Treating Radioactive	Wastes from Reprocessing Installations*	
Results and Developments on the Disposal of Radioactive	Wastes in the Asse Salt Mine* ©Recent	
©Nuclear	Wastes in the Deep Seabed*	
Law Concerning the Disposal ©System Study, Radioactive	Wastes in the Federal Republic of Germany. Vol 4. The	
Republic of Germany: Accumulation of Radioactive	Wastes in the Federal Republic of Germany* the Federal	
of Radioactive Wastes in the Federal ©Radioactive	Wastes in the Federal Republic of Germany: Accumulation	
With, Underground Storage of Liquid Radioactive	Wastes in the USSR*Scientific Basis for, and Experience	
©The Feasibility of Incorporating Radioactive	Wastes in Asphalt or Polyethylene*	
©Fixation of Radioactive	Wastes in Clay-Flux Mixes*	
©Heat Experiments for Self Fixation of Radioactive	Wastes in Clay-Flux Mixtures*	
©The Feasibility of Storing Radioactive	Wastes in Columbia River Basalts*	
Analysis of the Fixation of High-Level Radioactive	Wastes in Concrete*Preliminary Engineering and Economic	
©Terminal Storage of Radioactive	Wastes in Geologic Formations*	
for Repositories of Solid High-Level and Alpha-Bearing	Wastes in Geological Formations* ©Site Selection	
©Disposal of Long-Lived Radioactive	Wastes in Italy*	
©Disposal of Radioactive	Wastes in Salt Domes*	
©The Migration of Long-Lived Radioactive Processing	Wastes in Selected Rocks*	
Appalachian ©Possibilities for Disposal of Industrial	Wastes in Subsurface Rocks or North Flank of	
of the Disposal of High Activity Solidified	Wastes in Underground Salt Mines*Vault: A Demonstration	
©Evaluation on the Disposal of Radioactive	Wastes into the North Pacific*	
of Solidification and Burial Methods for Radioactive	Wastes of Fuel Cycle* ©Development	
©Light-Water Reactor	Wastes to be Isolated in Geologic Formations*	
©Processing of High-Level Atomic	Wastes with the View to Ultimate Disposal*	
Democratic Republic* ©Disposal of Radioactive	Wastes Produced in Nuclear Installations in the German	
Pilot Plant (©Potential Microbial Impact on Transuranic	Wastes Under Conditions Expected in the Waste Isolation	
Pilot Plant (©Potential Microbial Impact on Transuranic	Wastes Under Conditions Expected in the Waste Isolation	
Pilot Plant (©Potential Microbial Impact on Transuranic	Wastes Under Conditions Expected in the Waste Isolation	
* ©Volume Reduction of Liquid Radioactive	Wastes Using Mechanically Agitated Thin-Film Evaporators	
©Underground Storage Condition for Radioactive	Wastes*	
©Degradation of BAP	Wastes*	
©Retrievable Surface Storage of High-Level Radioactive	Wastes*	
©Potential Microbial Impact on Transuranic	Wastes*	
©Geochemical Behavior of Long-Lived Radioactive	Wastes*	
©The Management of Canada's Nuclear	Wastes*	
©Management of Radioactive	Wastes*	
©Radioactive	Wastes*	
©Radioactive	Wastes*	
©Deep Disposal Systems for Radioactive	Wastes*	
©Potential of Denver Basin for Disposal of Liquid	Wastes*	
©Environmental Protection Requirements for Radioactive	Wastes*	
©Sea Disposal of Packaged Radioactive	Wastes*	
©Characteristics of	Wastes*	
©Vitrification of High Level ICPP Calcined	Wastes*	
©The Management and Storage of Commercial Power Reactor	Wastes*	

*Solid Forms for Savannah River Plant Radioactive	Wastes*	000643
*Miscellaneous Radioactive and Nonradioactive	Wastes*	000635
*Fission Fuel Cycle Solid Radioactive	Wastes*	000697
S. Program for the Immobilization of High-Level Nuclear	Wastes*	000586
Brookhaven Experience with Sea Disposal of Radioactive	Wastes*	000598
Ground Storage Facilities for Vitrified Highly Active	Wastes*	000573
Relations in Injection Disposal of Radioactive	Wastes*	000266
of Containers for Storing Solidified Radioactive	Wastes*	000095
Criteria for Repositories for High-Level Radioactive	Wastes*	000435
Gas Production during Long-Term Storage of Nuclear	Wastes*	000577
and Storage of Hanford's High-Level Radioactive Liquid	Wastes*	000669
Radiation Stability of Glasses for Highly Radioactive	Wastes*	000608
and Leachability of Vitrified Radioactive	Wastes*	000659
for Managing Post-Light-Water Reactor Nuclear	Wastes*	000519
for Storage and Disposal of Spent Fuel and High-Level	Wastes*	000329
to Minimize, Segregate, and Account for Alpha-Bearing	Wastes*	000610
Factors for Locating Repositories for Radioactive	Wastes*	000427
International Burial Area for High-Level Radioactive	Wastes*	000352
for Potential Underground Repositories of Radioactive	Wastes*	000018
for Potential Underground Repositories of Radioactive	Wastes*	000280
for Long-Term Storage of Highly Active Radioactive	Wastes*	000128
Storage of Bitumen Blocks Containing Radioactive	Wastes*	000581
for High-Level Idaho Chemical Processing Plant Zirconia	Wastes*	000628
on the Relative Hazard of Deep Burial of Nuclear	Wastes*	000084
Vol 4. The Law Concerning the Disposal of Radioactive	Wastes*	000495
Control at the Stage of Transporting Solid Radioactive	Wastes*	000567
Favorable to Deep Placement of High-Level Radioactive	Wastes*	000068
Salt Domes for the Storage or Disposal of Radioactive	Wastes*	000428
Bearing Horizons for Disposal of Liquid Radioactive	Wastes*	Prerequisites and Practice for Using Deep Water- 000203
of Geologic Disposal of Commercial Radioactive	Wastes*	United States Program for the Safety Assessment 000319
to the Problem of Geological Storage of Nuclear	Wastes*	Paths for Okla Reactor Products and Applications 000236
Western Oklahoma for Underground Storage of Radioactive	Wastes*	Permian Salt Deposits in the Texas Panhandle and 000417
*Vitrification of High Sodium-Aluminum	Wastes:	Composition Ranges and Properties* 000104
Prototype Sites of Repositories for Radioactive	Wastes:	Flood Basalt* *Hypothetical 000444
*PIRE	Wastes:	Permanent Isolation in Rock-Forming Minerals* 000149
Pertinent to the Migration of Radionuclides in Ground	Water	at the Nevada Test Site - Part 2: Annotated 000163
*Geochemistry of Ground	Water	at the Savannah River Plant* 000180
in Vertical Boreholes and Brine Invasion into Fresh	Water	Aquifer. 1. Numerical Study of Convective- 000273
in Vertical Boreholes and Brine Invasion into Fresh	Water	Aquifer. 2. Experimental Study of Convective- 000274
in Vertical Boreholes and Brine Invasion into Fresh	Water	Aquifer. 3. Numerical Simulation of Brine 000275
3. Numerical Simulation of Brine Invasion of Fresh	Water	Aquifers*Brine Invasion into Fresh Water Aquifer. 000275
U.S. Atomic Energy Commission, *Geology and Ground-	Water	Characteristics of the Hanford Reservation of the 000189
Waste *Evaluation of Isotope Migration-Land Burial	Water	Chemistry at Commercially Low-Level Radioactive 000167
*Evaluation of Isotope Migration-Land Burial	Water	Chemistry at Commercially Operated Low-Level 000166
*Evaluation of Isotope Migration-Land Burial	Water	Chemistry at Commercially Operated Low-Level 000168
Behavior of Radioactive Ions in Unsaturated Soils after	Water	Disposal into a Limited Surface, Part 2*of Soils: 000292
*Kinetic Model for Silicate Rock-Ground	Water	Interaction* 000008
Isolation in Geologic Formations. Volume 27. Ground	Water	Movement and Nuclide Transport* Radioactive Waste 000237
"As Low As Practicable" Guides - Fabrication of Light-	Water	Reactor Fuel from Enriched Uranium Dioxide* 000652
*Neutron and Gamma-Ray Sources in Light	Water	Reactor High-Level Nuclear Waste* 000540
*The Alternatives for Managing Post-Light-	Water	Reactor Nuclear Wastes* 000519
of Non-Fuel Bearing Hardware Resulting from Light	Water	Reactor Operations* Transportation, and Disposal 000651
Loss at Sea* *Evaluation of the Consequences of Light	Water	Reactor Spent Fuel and Plutonium Shipping Package 000118
Formations* *Light-	Water	Reactor Wastes to be Isolated in Geologic 000146
*Sources of Radioactive Waste from Light-	Water	Reactors and Physical and Chemical Properties* 000691
High Level Waste Canister SS-9 While in Air and	Water	Storage* *History of Prototype 000[illegible]
from Highly Radioactive Glass Blocks Buried Below the	Water	Table: Fifteen Years of Results* of Radioactivity 000685
*Determination of Transuranics in Environmental	Water*	000600
Removal of Cesium and Strontium from Fuel Storage Basin	Water*	000607
Migration in Rock and Porous Media, Penetrated by	Water*	*Calculations of Nuclide 000262
the Idaho Chemical Processing Plant Fuel Storage Basin	Water*	*Primary Source of Radionuclides Leaking into 000188
*Fundamental Prerequisites and Practice for Using Deep	Water-Bearing	Horizons for Disposal of Liquid 000203
Reactivity of Simulated Nuclear Waste Forms and	Water-Catalyzed	Waste-Rock Interactions* *Hydrothermal 000058
of the Mesozoic Aquifer System Below the Zone of Fresh	Water, Atlantic	Coastal Plain, North Carolina through 000366
Aspect for their Exploitation for Ultimate *The	Waterproof	Geometry of Salt Domes, An Important Safety 000407
of Sediment and Contaminant Transport in Surface	Waters*	1977 to September 1978 Mathematical Simulation 000289
Detection in Crystalline Rock Using Ultrasonic Shear	Waves*	*Fracture 000481
*Let's Go Slowly on a Nuclear Power Program Until	We've	Solved Waste Problems* 000527
*Oceanic Distribution of Radionuclides from Nuclear	Weapons	Testing* 000230
Sorption on Two United States Subsoils from Differing	Weathering	Intensity Areas* 237, and Technetium 99 000293
237, and Americium 241 on Two Subsoils from Differing	Weathering	Intensity Areas* of Technetium 99, Neptunium 000294
1. Numerical Study of Convective-Diffusive Flow in	Wellbores*	and Brine Invasion into Fresh Water Aquifer. 000273
2. Experimental Study of Convective-Diffusive Flow in	Wellbores*	and Brine Invasion into Fresh Water Aquifer. 000274
Fuel Services, Incorporated, Reprocessing Plant at	West Valley, New York*	to the Closing of the Nuclear 000169
Test Trench at Western New York Nuclear Service Center,	West Valley, New York*	Analysis of Soil Samples from 000173
Report for Public Comment*	Western New York	Nuclear Service Center Study, Final 000211
Analysis of Soil Samples from Test Trench at	Western New York	Nuclear Service Center, West Valley, 000173
of Permian Salt Deposits in the Texas Panhandle and	Western Oklahoma	for Underground Storage of Radioactive 000417
*History of Prototype High Level Waste Canister SS-9	While in Air	and Water Storage* 000101
of the BOL Data Structure in Consequence Analysis for	WIPP Repository*	*The Utilization 000223
*Radionuclide Migration Studies Associated with the	WIPP Site	in Southern New Mexico* 000280
Mexico. Special Studies of Waste Isolation Pilot Plant (	WIPP) Bonks*	Pilot Plant (WIPP) Site, Southeastern New 000073
Associated with the Waste Isolation Pilot Plant (	WIPP) Site	in New Mexico*of Radionuclides with Geomedia 000201
Characterization Report, Waste Isolation Pilot Plant (	WIPP) Site,	Southeastern New Mexico. Geochemistry* 000078
Characterization Report, Waste Isolation Pilot Plant (	WIPP) Site,	Southeastern New Mexico. Hydrology* 000456
Characterization Report, Waste Isolation Pilot Plant (	WIPP) Site,	Southeastern New Mexico. Regional Geology* 000453
Characterization Report, Waste Isolation Pilot Plant (	WIPP) Site,	Southeastern New Mexico. Seismology* 000455
Characterization Report, Waste Isolation Pilot Plant (	WIPP) Site,	Southeastern New Mexico. Site Geology* 000458

[illegible]

GEOGRAPHIC LOCATION INDEX

*** A ***

Africa, Gabon, Oklo 220, 236, 247, 538

Aiken 236

Alabama (W) 830

Alabama, Clarke County 271

Alaska, Amchitka Island 377

Antarctica 881, 291, 892

Appalachian Region 825

Atlantic Coastal Plain 365

Atlantic Ocean (B) 261

*** B ***

Belgium 536

*** C ***

California, Eureka, Humboldt Bay Reactor 280

California, La Jolla, La Jolla Beach 235

California, Pleasanton, Vallecitos Nuclear Center 135

California, San Clemente, San Onofre Reactor 280

Canada 329, 870, 873

Canada, Ontario, Chalk River 685

Canada, Ontario, Elliot Lake, Denison Mine 889

Canada, Ontario, George MacLeod Mine 889

Canada, Ontario, Manitouwadge, Geco Mine 889

Canada, Ontario, Sudbury, Creighton Mine 889

Canadian Shield 318, 825

Colorado, Golden, Rocky Flats Plant 136

Connecticut, Haddam Neck, Connecticut Yankee Reactor 280

*** D ***

Denver Basin 800, 801

*** E ***

Eleana Formation, Unit J, Hole UE17E 396

England 688

*** F ***

Finland 536

France 535, 688

*** G ***

Georgia 265

Georgia, Jefferson Davis County 271

Georgia, Ware County 271

German Federal Republic 536, 688

German Federal Republic, Asse Site 177, 397

Germany, Karlsruhe, Kernforschungszentrum Karlsruhe 612

Greenland 491

Gulf Coast Basin 529

Gulf Coast Region 825

*** I ***

Idaho, Idaho Falls, Idaho National Engineering Laboratory 168, 188, 209, 210, 308, 309, 380, 381, 383, 384, 385, 475, 525, 583, 692, 693

Idaho, Idaho Falls, Idaho National Engineering Laboratory, Idaho Chemical Processing Plant 116, 117, 138, 208, 598

Idaho, Idaho Falls, Idaho National Engineering Laboratory, Lemhi Range 207, 382

Idaho, Idaho Falls, Idaho National Engineering Laboratory, Radioactive Waste Management Complex 176, 207, 208, 307, 382

Idaho, Idaho Falls, Idaho National Engineering Laboratory, Site 18 207, 382

Idaho, Idaho Falls, Idaho National Engineering Laboratory, Subsurface Disposal Area 181

Idaho, Idaho Falls, National Reactor Testing Station 192, 418, 671, 687

Idaho, Idaho National Engineering Laboratory, Idaho Chemical Processing Plant 158, 386

Illinois, Argonne, Argonne National Laboratory 31, 600

Illinois, Morris Dresden-1 Reactor 280

Indiana, Crawford County 393

Indiana, Martin County, Crane Naval Ammunition Depot 393

Indiana, Perry County 393

Italy 536

Italy, Matera, Trisaia Center 403

*** K ***

Kansas 879

Kansas, Hutchinson 359

Kansas, Hutchinson, Carey Salt Mine 350

Kansas, Lincoln County, Lyons 826

Kansas, Lyons, Lyons Mine 12, 13, 51, 55, 109, 316, 328, 361, 395, 530, 531, 533, 539

Kansas, Salina Basin 396, 400

Kentucky, Carnforon 370

Kentucky, Nasville 370

Kentucky, Maxey Flats 166, 167, 168

*** L ***

Lucann Salt Dome 398

Louisiana 313

Louisiana, Iberia Parish, Avery Island Mine 61

Louisiana, Iberia Parish, Jefferson Island Mine 61

Louisiana, Salt Dome Basin 685

Louisiana, Salt Dome Basin, Webster and Bienville Parishes 820, 828, 861, 872

*** M ***

Massachusetts, Rowe, Yankee Reactor 280

Michigan Basin (SE) 368, 400

Michigan, Charlevoix, Big Rock Point Reactor 280

Michigan, Hancock, Centennial Mine 889

Michigan, Ontonagon County, White Pine Copper Mine 889

Minnesota, Monticello, Monticello Nuclear Generating Plant 595

Minnesota, Tower, Soudan Mine 889

Mississippi 313

Mississippi (W) 830

Missouri, Kansas City, Bethany Falls 371

*** N ***

Netherlands 263

GEOGRAPHIC LOCATION INDEX

Nevada, Clark County, Virgin River, Lake Mead 439

Nevada, Engineer Maintenance and Disassembly Facility 135

Nevada, Nevada Test Site 28, 83, 88, 163, 305, 313, 423, 446, 447

Nevada, Nevada Test Site, Area 17 88

Nevada, Nevada Test Site, Jackass Flats 419

New Jersey, Toms River, Oyster Creek Reactor 280

New Mexico 72, 300, 426

New Mexico, Eddy County 431

New Mexico, Lea County 431

New Mexico, Los Alamos, Los Alamos Scientific Laboratory 135, 171, 419

New Mexico, Los Alamos, Los Alamos Scientific Laboratory, Burial Site TA-54, Area C 7, 8

New Mexico, Los Alamos, Los Alamos Scientific Laboratory, DP Canyon 283

New Mexico, Los Alamos, Los Alamos Scientific Laboratory, Technical Area 54, Area G 196

New Mexico, Los Medanos 259

New Mexico, Los Medanos Area, WIPP Site 241, 362

New Mexico, Waste Isolation Pilot Plant 73, 74, 338, 453, 454, 455, 456

New Mexico, WIPP Site 137, 431

New York 213

New York, Appalachian Basin, North Plank 400, 429

New York, Ashford, Nuclear Service Center 633

New York, Buchanan, Indian Point-1 Reactor 280

New York, Cattaraugus County, West Valley 211, 589

New York, Cattaraugus County, West Valley, Western New York Nuclear Service Center 173

New York, Oswego, Nine Mile Point Reactor 280

New York, Rochester, Robert E. Ginna Reactor 280

North Carolina, Bladen County 271

*** O ***

Ohio 313

Ohio (NE) 425

Ohio, Barberton, Barberton Mine 371

Ohio, Columbus, Battelle Columbus Laboratories Hot Cell Facility 135

Ohio, Irvoton, Alpha Portland Cement Mine 370

Ohio, Lewisburg, Lewisburg Mine 370

Ohio, Zanesville, Jonathan Mine 370

Oklahoma (N) 417

*** P ***

Pacific Ocean (N) 303

Pacific Ocean, Kuroshio Region 475

Paradox Basin 384, 425, 528

Pennsylvania, Appalachian Mountains (C), Dunning Cove Syncline 400, 408

Pennsylvania, Peach Bottom, Peach Bottom-1 Reactor 280

Pennsylvania, Shippingport, Shippingport Reactor 280

Permian Basin 425

*** R ***

Rocky Mountains 425

*** S ***

Salina Basin 384, 425, 528

San Juan Basin 400, 443

Savannah River Plant 206

South Carolina 206, 293, 365

South Carolina, Aiken, Savannah River Laboratory 180, 271, 615, 666, 683

South Carolina, Aiken, Savannah River Plant 4, 99, 478, 603

South Carolina, Barnwell County 94

South Carolina, Barnwell, Barnwell Nuclear Fuel Plant 205

Spain 536

Supai Basin 425

Sweden 123, 262, 536

Sweden, Stripa 56, 350, 382, 398, 422, 487

Sweden, Stripa, Stripa Mine 378, 381, 397, 437, 538, 535

Switzerland 536

*** T ***

Tennessee (N) 430

Tennessee, Oak Ridge 76, 87, 421

Tennessee, Oak Ridge, Oak Ridge National Laboratory 178, 579, 642

Tennessee, Oak Ridge, Oak Ridge National Laboratory, Shale Fracturing Facility 213

Texas 313

Texas Panhandle 417

Texas, Anderson County 353, 412, 432, 438

Texas, Cherokee County 412, 432, 438

Texas, Henderson County 412, 432, 438

Texas, Smith County 353, 412, 432, 438

Texas, Van Zandt County 412, 432, 438

Texas, Wood County 412, 432, 438

*** U ***

United Kingdom 375

United States 536

USSR 573, 624

Utah 313

Utah, Grand County, Salt Valley Anticline 411

*** V ***

Virginia, Lynchburg, Lynchburg Research Center 135

*** W ***

Washington (N) 293

Washington, Gable Mountain 52

Washington, Hanford Reservation 212, 268, 290, 313, 630

Washington, Pasco Basin 388, 389

Washington, Pasco Basin, Saddle Mountains 387

Washington, Pasco Basin, Saddle Mountains, Ostrander Ridge 390

Washington, Richland, Battelle-Pacific Northwest Laboratories 608, 661

Washington, Richland, Hanford Atomic Products Operation 183, 191

Washington, Richland, Hanford Engineering Development Laboratory 135

Washington, Richland, Hanford Reservation 26, 179, 197, 221, 222, 286, 387, 406, 808, 851, 669

GEOGRAPHIC LOCATION INDEX

Washington, Richland, Hanford Reservation, 241-Bx Tank Farm 195

Williston Basin 825

Wisconsin, Genoa, La Crosse Reactor 280

*** A ***

Activation Energy 679

Alkalinity 213

Aquifer Thickness 365

*** B ***

Bulk Modulus 23

*** C ***

Calcium Carbonate Content 183, 193, 194, 195, 298, 397, 403

Clay Content 97, 183, 298, 403

Compressional Wave Velocity 73

Compressive Strength 21, 22, 23, 59, 60, 384, 385, 390, 655

Corrosion Rate 102, 103, 129, 132, 151

Creep Rate 73

*** D ***

Density 53, 59, 60, 66, 73, 204, 291, 302, 357, 385, 403, 570, 580, 597, 599, 602, 636

Depth to Water Table 189, 192, 193, 194, 195, 200, 366, 418, 479, 581

Diffusion Coefficient 128, 218, 291

Diffusivity 360

Distribution 198

Distribution Coefficient 28, 45, 183, 191, 240, 288, 291, 293, 294, 295, 300, 302, 377, 447, 576, 623, 638

*** E ***

Eh 11, 38

Elastic Modulus 837

*** F ***

Flow Rate 41, 204, 298, 645, 649, 670

*** G ***

Grain Size 198

Grain Size Distribution 29, 191, 193, 195, 301, 305, 403, 415

*** H ***

Heat Capacity 21, 23, 76, 384

Heat of Vaporization 604

Hydraulic Conductivity 302, 366, 406, 431, 448

Hydraulic Gradient 192, 200, 291, 302, 448

Hydraulic Head 444, 456

Hydraulic Velocity 90, 268, 431, 456, 645

*** I ***

Injection Pressure 218

Ion Exchange Capacity 2, 27, 29, 64, 165, 178, 183, 268, 288, 403, 572, 576, 597, 623, 675, 678

*** L ***

Leaching Rate 99, 101, 128, 153, 155, 216, 229, 302, 358, 447, 467, 519, 570, 574, 596, 620, 628, 634, 645, 655, 659, 664, 673, 689

*** M ***

Mass Action Coefficient 572

Mean Precipitation 456, 581

Migration Rate 249, 252, 298

Moisture Content 59, 173, 229, 265, 403, 437

*** P ***

Percent Adsorption 33, 250

Permeability 22, 23, 29, 49, 59, 69, 77, 97, 156, 189, 364, 366, 400, 403, 447

Permeability Coefficient 173, 291

pH 11, 28, 29, 38, 42, 45, 60, 78, 103, 166, 168, 178, 180, 183, 191, 296, 299, 298, 302, 397

Plasticity 173, 359, 403

Poisson's Ratio 21, 22, 23, 384, 469

Porosity 21, 22, 23, 29, 53, 60, 97, 189, 191, 291, 302, 364, 366, 377, 384, 389, 400, 403, 408, 431, 448, 570, 602, 685, 689

Pressure (Vapor) 16

*** R ***

Receptivity 368, 400

Resistivity 73, 656

River Flow Rate 291

*** S ***

Saturation Capacity 183

Shear Modulus 23

Solubility (of CaSO4 in two brine solutions) 16

Sorption Coefficient 73, 241, 251, 252

Specific Activity 213

Specific Conductance 166, 168

Storage Coefficient 192, 448

Stratigraphic Unit Thickness 180, 189, 192, 193, 194, 195, 291, 305, 365, 366, 393, 406, 410, 418, 453, 454, 478, 579, 581

*** T ***

Temperature 16, 40, 78, 103, 166, 168, 360, 377, 381, 411, 447, 568, 602, 623

Tensile Strength 21, 22, 23, 384, 389

Thermal Conductivity 21, 23, 53, 59, 60, 61, 63, 73, 76, 359, 360 384, 385, 423, 467, 569, 602

Thermal Expansion 21, 22, 23, 69, 76, 99, 384, 447

Thermal Loss 75

Total Dissolved Solids 397

Total Ion Concentration 227, 397, 446, 607

Transmissivity 189, 192, 268, 377, 431

*** V ***

Viscosity 656, 657

Volumetric Water Content 21, 384

*** W ***

Waste Volume 1, 30, 179, 181, 184, 191, 192, 200, 201, 213, 295, 351, 476, 482, 580, 584, 587, 593, 621, 625, 627, 633, 638, 682, 698

*** Y ***

Young's Modulus 21, 22, 23, 384, 469

MEASURED RADIONUCLIDES INDEX

*** A ***

Ag 15

Am 241 28, 128, 168, 181, 204, 230, 237, 249, 250, 252, 293, 298, 300, 301, 302, 599

Am 243 73, 241

*** B ***

Ba 133 28

*** C ***

C 14 172

Ca 45 85

Ce 103, 286

Ce 141 28

Ce 144 73, 101, 200, 227, 241, 300, 358, 566, 620, 636

Cf 252 133

Cm 15, 214

Cm 242 133, 288, 302, 358

Cm 243 300

Cm 244 73, 99, 128, 237, 241, 301, 302, 358

Co 60 168, 175, 179, 191, 197, 204, 285, 302, 358, 584

Cr 51 302

Cs 27, 29, 33, 214, 604, 660

Cs 134 204, 297, 302, 358

Cs 135 123

Cs 137 28, 45, 60, 73, 101, 168, 175, 178, 179, 183, 191, 197, 204, 230, 237, 241, 285, 300, 302, 358, 377, 467, 486, 591, 584, 620, 627, 636, 638, 655, 659, 664

*** E ***

Eu 152 28, 73, 175, 241, 300

Eu 154 175, 204, 237, 358, 620, 636

Eu 155 175, 204, 358

*** F ***

Fe 55 230, 302

*** G ***

G 51 584

Gd 153 73, 241

*** H ***

H 3 166, 172, 180, 197, 221, 292, 298, 302, 377, 476, 584, 682

*** I ***

I 32

I 129 9, 250, 302

I 131 73, 200, 241, 250, 292, 297, 584

*** K ***

K 42 85

Kr 85 172

*** M ***

Mn 54 302, 358

*** N ***

Nb 95 200, 285

Ni 59 302

Np 15, 32

Np 237 11, 123, 153, 204, 250, 293, 298, 300, 301, 302

*** O ***

O 18 190

*** P ***

Pa 233 300

Po 210 235

Pr 144 200

Pu 15, 32, 170, 183, 197, 214, 216, 230, 467, 476, 486, 636, 682

Pu 237 28, 249, 250, 252, 300

Pu 238 73, 99, 128, 168, 181, 204, 237, 241, 249, 255, 271, 288, 295, 302, 599, 623, 638

Pu 239 9, 91, 153, 181, 201, 204, 235, 237, 250, 295, 302, 358, 599, 627

Pu 239, 240 168, 271

Pu 240 204, 235, 237, 358, 599

Pu 241 204, 302, 588, 599

*** R ***

Ra 226 123

Rare Earths 178

Rh 106 285, 358

Ru 604, 660

Ru 106 73, 162, 178, 191, 192, 197, 198, 221, 241, 285, 300, 584, 620, 636

*** S ***

S 35 180

Sb 125 73, 175, 204, 241, 300, 358, 620

Se 79 594

Sm 151 237

Sr 27, 214, 660

Sr 85 28, 73, 241, 292, 298, 300, 664

Sr 90 85, 60, 168, 178, 179, 183, 191, 197, 200, 204, 227, 229, 230, 237, 286, 302, 358, 467, 486, 591, 620, 627, 636, 638, 682, 685, 659

*** T ***

Tc 32

Tc 95m 28, 300

Tc 99 4, 11, 73, 221, 237, 241, 250, 293, 298, 302, 598, 627, 636

*** U ***

U 15, 123, 377, 476, 660, 682

U 233 251, 623

U 234 204

U 235 204, 300

U 236 204

U 237 28, 251

U 238 204

*** Y ***

Y 183

Y 90 229

*** Z ***

Zn 65 175, 584

Zr 95 200, 285

PUBLICATION DESCRIPTION INDEX

*** A ***

A/CONF.15/P-1255 415

A/CONF.15/P-2310 85

A/CONF.15/P-2431 489

Advances in Chemistry Series 153 31, 551, 575, 579, 582, 589, 600, 648, 669, 671, 683, 684, 687

AE-RT-77-3264 262

AEC-TR-2867 272

AECL-5597 329

AERE-R 8374 416

American Association of Petroleum Geologists Bulletin 59(5):921 452

American Association of Petroleum Geologists, Memoir 10 368, 371, 398, 399, 400, 401, 408, 425, 483, 480

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American Mineralogist 51:1649-1656 23

American Mineralogist 51:324-335 27

American Nuclear Society Annual Meeting, Atlanta, GA, June 5, 1979, (8 pp.) 503

American Nuclear Society Annual Meeting, New York, NY, June 12-16, 1977, (p. 297) 230

American Nuclear Society Annual Meeting, New York, NY, June 12-16, 1977, (pp. 247-248) 127

American Nuclear Society Annual Meeting, New York, NY, June 12-17, 1977, (pp. 297-298) 304

American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (p. 368) 188

American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (p. 571) 508

American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (pp. 81-82) 538

American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (pp. 82-83) 529

American Nuclear Society Annual Meeting, San Diego, CA, June 18-22, 1978, (18 pp.) 496

American Nuclear Society Meeting, Philadelphia, PA, June 23-28, 1978, (18 pp.) 498

American Nuclear Society Spring Meeting, Atlanta, GA, June 3-8, 1979, (3 pp.) 506

American Nuclear Society Winter Meeting, San Francisco, CA, November 11-15, 1973 119

American Nuclear Society Winter Meeting, San Francisco, CA, November 11-15, 1973, (1 p.) 158

American Nuclear Society Winter Meeting, Washington, DC, October 27-31, 1974, (pp. 204-206) 703

An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management, Vol. 2, English, T.D. (Project Manager), Appendix B, (pp. B-1 - B-63), 235 pp. 324

An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management Vol. 2, English, T.D. (Project Manager), Appendix D, (pp. D-1 - D-54), 235 pp. 500

An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management, Vol. 2, English, T.D. (Project Manager), Appendix G, (pp. G-1 - G-6), 235 pp. 700

An Analysis of the Back-End of the Nuclear Fuel Cycle with Emphasis on High-Level Waste Management, Vol. 2, English, T.D. (Project Manager), Appendix C, (pp. C-1 - C-62), 235 pp. 325

An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 21-94), 475 pp. 472

An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 211-352), 475 pp. 485

An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 353-426), 475 pp. 420

An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, (pp. 95-210), 475 pp. 461

ANL-75-64 252

ANL-76-22 251

ANL-76-46 250

ANL-76-68 297, 298, 336

ANL-78-8 299

ANL-8152 672

ANL/RS-26 (Part 1) 253

Annals of Nuclear Energy 3:285-295 327

Annals of Nuclear Energy 4:577-528 159

Annual Scientific Report 1975, (pp. 8-18 - 8-19), 217 pp. 682

Annual Scientific Report 1976, (pp. 8-12 - 8-14), 168 pp. 676

ARH-LD-126 193

ARH-LD-129 194

ARH-LD-130 195

ARH-LD-162 221

ARH-SA-232 290

ARH-SA-292 222

ARH-ST-111 129

ARH-1278 170

ARH-2874 162

Atom 240:250-253 613

Atom 257:74-76 317

Atom 261:186-188 375

Atomnaya Energiya 22(5):397-399 626

Atomnaya Energiya 26(5):411-419 624

Atomwirtschaft Atomtechnik 22(12): 601-602 497

Aware, October 1976:6-11 587

*** B ***

BLG-515/76 682

BLG-520/77 476

BNL-NUREG-50695 168

BNL-NUREG-50739 167

BNL-NUREG-50855 166

BNL-50620 601

BNWL-SA-5572 580

BNWL-tr-189 573

BNWL-tr-273 291

BNWL-1868 197

BNWL-1814 216

BNWL-1889 298

BNWL-1927 273

BNWL-1970 225, 226, 232, 234

BNWL-2019 701

BNWL-2111 608

BNWL-2126 112, 175, 318, 333

BNWL-2127 215

BNWL-2234 630

BNWL-36 (Part 4) 2

Bulletin of the Atomic Scientists 29:4-9, 50-52 492

Bulletin of the International Association of Engineering Geology (18):173-178 352

*** C ***

California Geology, p. 82 445

CA-35402 126

Chemical Engineering Division Fuel Cycle Programs Progress Report, January-March 1978, R.J. Steindler, et al., Section 7, (pp. 176-182), 202 pp. 298

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Chemical Engineering Programs, April 1976:63-64 603

CONF-721060 448

CONF-721101 119, 158

CONF-740346 21, 551, 575, 579, 582, 589, 600, 644, 669, 671, 683, 686, 687

CONF-740608 498

CONF-741017 703

CONF-750411 247, 281, 463, 518

CONF-750522 356

CONF-751044 509

CONF-751105 290

CONF-760310 97, 177, 329, 403, 407, 450, 519, 523, 574, 586, 607, 608, 618, 645, 653

CONF-760701 125, 528, 613

CONF-760744 425

CONF-760813 136

CONF-761020 220, 424, 482

CONF-761038 580

CONF-761175 263

CONF-761236 373

CONF-770301 486

CONF-770303 148

CONF-770383 273, 274, 275, 348, 460, 462, 478

CONF-770565 291, 319

CONF-770611 127, 230

CONF-770749 6, 14, 32, 36, 57, 79, 82, 85, 87, 88, 89, 137, 145, 406, 421, 423, 482, 446

CONF-771109 308

CONF-771208 419

CONF-780136 386

CONF-780316 268

CONF-780568 86

CONF-780672 188, 456, 504, 529, 538

CONF-780819 150, 598

CONF-781121 1, 3, 10, 16, 17, 39, 40, 54, 62, 71, 72, 77, 78, 80, 81, 93, 102, 104, 105, 110, 111, 113, 115, 118, 122, 124, 126, 130, 131, 132, 138, 141, 145, 151, 153, 155, 156, 157, 214, 223, 228, 231, 240, 244, 246, 248, 257, 258, 270, 276, 278, 287, 330, 354, 464, 494, 517, 560, 562, 569, 591, 597, 605, 611, 614, 621, 627, 629, 635, 637, 640, 658, 662, 663, 664, 665, 673, 686, 689

CONF-781241 103

CONF-790602 320, 503, 565, 586

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 10 (pp. 10-1 - 10-21), 837 pp. 558

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 11, (pp. 11.1 - 11.8), 837 pp. 559

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 2, (pp. 2-1 - 2-73), 837 pp. 850

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 3, (pp. 3-1 - 3-36), 837 pp. 146

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 4, (pp. 4-1 - 4-109), 837 pp. 553

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 5, (pp. 5-1 - 5-52), 837 pp. 554

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 7, (pp. 7-1 - 7-4), 837 pp. 555

Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations, Ch. 8, (pp. 8-1 - 8-7), 837 pp. 556

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Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing Uranium 233 and Thorium, Section 5, 8, (pp. 64-67), 102 pp. 695

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DOE/ET-0075 705

DOE/ET-0081 207, 208, 209, 210, 307, 308, 309, 340, 341, 342, 343, 344, 345, 475, 525

DOE/ET-0004 152

DOE/SR-NE-79-1 206

DOE/DP-76/089a 267

DP-RS-76-39 125

DP-RS-76-41 577

DP-RS-76-66 95

DP-RS-76-69 406

DP-RS-77-23 615

DP-RS-77-75X 4

DP-RS-78-9 133

DP-RS-79-2X 506

DP-1356 100

DP-1399 460

DP-1400 674

DP-1400 608

DP-1411 620

DP-1425 139

DP-1440 655

DP-1440 467

DP-1459 100

DP-1460 98

DP-1482 99

DP-1507 657

DP-1517 656

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Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 15, (pp. 15-1 - 15-4), 586 pp. 525

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Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 3, (pp. 3-1 - 3-36), 586 pp. 207

Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 4, (pp. 4-1 - 4-16), 586 pp. 208

Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 5, (pp. 5-1 - 5-12), 586 pp. 387

Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 6, (pp. 6-1 - 6-8), 586 pp. 209

Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 7, (pp. 7-1 - 7-23), 586 pp. 475

Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste, Section 8, (pp. 8-1 - 8-10), 586 pp. 210

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HRI-D-202 699

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IAEA-CN-36/313 802

IAEA-CN-36/345 283

IAEA-CN-36/350 392

IAEA-SM-199/87 290

IAEA-SM-207-81 574

IAEA-SM-207/19 607

IAEA-SM-207/20 608

IAEA-SM-207/42 407

IAEA-SM-207/44 450

IAEA-SM-207/46 618

IAEA-SM-207/47 523

IAEA-SM-207/48 97

IAEA-SM-207/5[illegible] 653

IAEA-SM-207/55 35

IAEA-SM-207/67 519

IAEA-SM-207/69 588

IAEA-SM-207/70 94

IAEA-SM-207/75 636

IAEA-SM-207/82 177

IAEA-SM-207/88 803

IAEA-SM-207/91 329

IAEA-SM-207/98 645

IAEA-SM-10/10 116

IAEA-SM-10/25 136

IAEA-TC-119/26 236

IAEA-TC-119/30 504

ICP-1071 116

ICP-1072 117

ICP-1087 660

ICP-1112 194

ICP-1118 631

ICP-1176 598

ICP-1177 602

ICP-1105 628

ICP-1106 154

ICP-1107 245

ICP-1109 204

IDO-22042-USGS 418

IEA-75-3 520

IEC-4460C0417 465

Isotoptechnika Budapest 20:34-47 593

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Jaderna Energie 22(9):343-349 322

Journal of Environmental Quality 7(3):422-428 181

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JPL Publication 77-59 320, 325, 500, 700

JPL Publication 77-59, Volume 2 321

*** K ***

K-76-1008(R) 563

KBS-TR-29 466

KFK-2255 625

KFK-2328 612

KFK-2329 202

*** L ***

LA-UR-76-2096 220

LA-UR-77-2686 419

LA-7220-T 69

LA-7250-M 134

LA-7308-MS 196

LA-7456-MS 28

LA-7476-PR 160

LA-7549-MS 171

LA-7563-MS 610

LA-7582-PR 86, 229

LA-7609-MS 83

LA-7669-PR 87

LA-7674-MS 91

LA-7700-PR 8

LA-7839-PR 7

LA-7910-PR 9

LA-8109-PR 65

LA-8230-MS 107

LBL-7004 380

LBL-7022 219

LBL-7038 53

LBL-7047 218

LBL-7049 407

LBL-7051 481

LBL-7062 56

LBL-7063 534

LBL-7071 414

LBL-7072 381

LBL-7073 18, 19, 352

LBL-7074 469

LBL-7079 398

LBL-7080 422

LBL-7081 350

LBL-7082 1/2 378

LBL-7086 535

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LBL-8285 397

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Mechanical and Thermal Design Considerations for Radioactive Waste Repositories in Hard Rock, (pp. 17-26), 26 pp. 382

Memoir 18 355, 632

MIT Research Report 75-28, Volumes 1 and 2 52

MIT Research Report 76-26 56

MLM-2320 623

MLM-2509 634

MLM-2517 570

MLM-2534 570

MLM-2569 571

MLM-2575 622

Modern Geology 6:31-36 441

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NAS-NRC Publication No. 519 471

NASA TM X-2911 430

NASA TM X-2912 433

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NTIS/PS-77/0661 367

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NUREG-0194 506

NUREG/CR-0131 326

NUREG/CR-0400 512

NUREG/CR-0420 92

NUREG/CR-0481 103

NUREG/CR-0495 543

NUREG/CR-0536 255

NUREG/CR-0644 173

NUREG/CR-0658 289

NUS-3020 493

NVO-0039-35 396

NVO-171 305

NVSGS/79-2401 173

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OLS-78-140 322

OLS-78-285 625

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ORNL-15 536

ORNL-17 15

ORNL-7 383

ORNL/Sub/79/E511-01100-10 65

ORNL-CY-78-7-82 12, 13, 51, 55, 109, 316, 328, 361, 395, 530, 531, 533, 539

ORNL-2415 178

ORNL-4555 (Draft) 12, 13, 51, 55, 109, 316, 328, 361, 395, 530, 531, 533, 539

ORNL-4751 599

ORNL-4928 243

ORNL-5050 42

ORNL-5226 323

ORNL-5362 147

ORNL-5524 59

ORNL-5526 81

ORNL/NUREG/TM-4 230, 500, 501, 502, 695

ORNL/NUREG/TM-7 521

ORNL/SUB-3706/1 174

ORNL/SUB-3706/6 67

ORNL/SUB-3960/2 52, 66

ORNL/SUB-4269/19 540

ORNL/SUB-4269/2 440

ORNL/SUB-4269/3 140

ORNL/SUB-4269/9 541

ORNL/SUB-75/38203/1 479

ORNL/TM-4401 254

ORNL/TM-4724 585

ORNL/TM-4902 652

ORNL/TM-5462 20

ORNL/TM-5976 507

ORNL/TM-6000 690

ORNL/TM-6162 120

ORNL/TM-6192 566

ORNL/TM-6573 60

ORNL/TM-6809 61

ORNL/TM-6863 485

ORNL/TM-6882 601

ORNL/TM-7052 63

ORNL/tr-4522 567

ORNL/tr-4523 187

ORNL/tr-4567 466

ORNL/tr-4638 5

ORNL/tr-4641 229

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PB-256 903 267

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PNL-SA-6766 70

PNL-2247 679

PNL-2278 101

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PNL-2444 564

PNL-2448 300

PNL-2625 661

PNL-2643 542

PNL-2664 350

PNL-2719 697

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PNL-2903 26

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Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 11, (pp. 11.1-11.229), 822 pp. 361

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Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, Ch. 2, (pp. 2.1-2.26), 822 pp. 13

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RFP-2469 43

RFP-2655 490

RFP-2730 349

RFP-2749 647

RFP-2783 616

RHO-BWI-78-100 APR 390

RHO-BWI-78-100 FEB 387

RHO-BWI-78-100 JAN 388

RHO-BWI-78-100 MAR 389

RHO-LD-77-3 D 335

RHO-SR-36 260

RHO-ST-14 198

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SAC-01 487

SAC-03 469

SAC-05 422

SAC-06 535

SAC-07 414

SAC-08 350

SAC-09 374

SAC-10 18, 19, 382

SAC-11 381

SAC-14 56

SAC-15 398

SAC-17 90

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SAND-76-5992 482

SAND-76-9137 148

SAND-77-0369 590

SAND-77-0723 568

SAND-77-1021 363

SAND-77-119.2C 86

SAND-77-1226C 386

SAND-77-1406 96

SAND-77-1529 37

SAND-77-1816 50

SAND-77-1872 183

SAND-78-0108 38

SAND-78-0297 241

SAND-78-0566 25

SAND-78-1122 532

SAND-78-1429 537

SAND-78-1596 73, 74, 334, 453, 454, 455, 456

SAND-78-2111C 103

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Science Focus 59:3-8 527

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Second Biennial Topical Meeting on the Ocean, Nuclear Energy and Man, Tarpon Springs, FL, May 7-9, 1975 356

Soil Science Society of America Journal 42:51-56 288

Southeastern Geology 21(1):75-82 427

Soviet Atomic Energy 22(1):496-498 626

Soviet Atomic Energy 26(1):469-478 624

Soviet Soil Science (9):936-938 227

SRA-4 495

STI/PUB/433 97, 177, 403, 407, 450, 519, 523, 574, 588, 607, 608, 618, 645, 653

Studies of Transuranic Waste Storage under Conditions Expected in the Waste Isolation Pilot Project (WIPP), July 1-September 30, 1978, (pp. 13-15) 16 pp. 224

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Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 165-185), 253 pp. 401

Subsurface Disposal in Geologic Basins - A Study of Reservoir Strata, J.E. Galley (Ed.). American Association of Petroleum Geologists, Tulsa, OK, (pp. 186-231), 253 pp. 483

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Summary Progress Report RSI-0002 178

*** T ***

Technical Memorandum Report RSI-0028 591

Technical Memorandum Report RSI-0083 540

TEI-791 379

TIT-28905-1 211

TIT-7517 (Part 1B) 609, 617, 682, 678, 698

TIT-7550 183, 693

TIT-7613 646

TIT-7628 286

TIT-7644 58, 572, 576, 606, 638, 689, 677

Transactions of the American Nuclear Society 17:326 158

Transactions of the American Nuclear Society 17:327 119

Transactions of the American Nuclear Society 18:194-195 498

Transactions of the American Nuclear Society 19:204-206 703

Transactions of the American Nuclear Society 20:682-684 405

Transactions of the American Nuclear Society 20:691-692 463

Transactions of the American Nuclear Society 20:695-696 247

Transactions of the American Nuclear Society 26:247-248 127

Transactions of the American Nuclear Society 26:297 230

Transactions of the American Nuclear Society 26:297-298 304

Transactions of the American Nuclear Society 28:364 184

Transactions of the American Nuclear Society 28:371 504

Transactions of the American Nuclear Society 28:81-82 538

Transactions of the American Nuclear Society 28:82-83 529

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TREE-1064 176

TREE-1172 187

TREE-1250 (Volume 2) 164, 583, 692, 693

TREE-1265 186

TREE-1277 650

TREE-1286 185

*** U ***

U.S. EPA Background Report 526

U.S. Geological Survey Circular 779 362

U.S. Geological Survey Professional Paper 1088 365

U.S. Geological Survey Professional Paper 712-B 377

U.S. Geological Survey Professional Paper 717 159

UCRL-50050-78 447

UCRL-51791 498

UCRL-52078 Part 2 163

UCRL-52199 84

UCRL-52304 49

UCRL-52604 48

UCRL-80841 269

UCRL-80058 505

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USGS OFR-77-761 811

USGS-OFR-77-123 831

USGS/WRD/WRI-76/056 267

*** W ***

WASH-275 331

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*** Y ***

Y-2161 76

Y/OWI-7 528

Y/OWI-9 313

Y/OWI/INT-3/9 312

Y/OWI/SUB-3785/4 68

Y/OWI/SUB-3785/6 444

Y/OWI/SUB-4112/37 420, 428, 461, 472, 485

Y/OWI/SUB-4251/2 370

PUBLICATION DESCRIPTION INDEX

T/CWI/SWB-4251/3 371

T/CWI/SWB-4367/1 489

T/CWI/SWB-4494/1 417

T/CWI/SWB-4496/2 484

T/CWI/SWB-7009/4 483

T/CWI/SWB-706212 393

T/CWI/SWB-76/16502 563

T/CWI/SWB-76/16509 561

T/CWI/SWB-76/16514 370

T/CWI/SWB-76/16531 480

T/CWI/SWB-76/87950 430

T/CWI/SWB-76/99939 353, 412, 432, 438

T/CWI/SWB-77/10260 6, 14, 32, 36, 57, 79, 82, 85, 87, 88, 89, 137, 149, 315, 406, 421, 423, 482, 486

T/CWI/SWB-77/16535/3 499

T/CWI/SWB-77/16537 273, 274, 275, 348, 460, 482, 478

T/CWI/SWB-77/42505 493

T/CWI/TH-14 427

T/CWI/TH-20 373

T/CWI/TH-36-23 296

T/CWI/TH-36/1 552

T/CWI/TH-36/10 547

T/CWI/TH-36/12 548

T/CWI/TH-36/14 549

T/CWI/TH-36/16 550

T/CWI/TH-36/18 546

T/CWI/TH-36/19 75

T/CWI/TH-36/20 24

T/CWI/TH-36/21 237

T/CWI/TH-36/3 385

T/CWI/TH-36/4 384

T/CWI/TH-36/5 21

T/CWI/TH-36/6 22

T/CWI/TH-36/7 23

T/CWI/TH-36/8 545

T/CWI/TH-42 121

T/CWI/TH-44 146, 457, 458, 553, 554, 555, 556, 557, 558, 559

T/CWI/TH-47 368

T/CWI/TH-52 529

T/CWI/TH-8 426