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FOREWORD

The goal of the U.S. nuclear waste management program is to provide assurance that existing and future nuclear waste from military and civilian activiies, including spent fuel from the once-through nuclear power cycle, can be isolated from the biosphere so as to pose no significant threat to public health and safety. The U.S. Department of Energy (DOE) is responsible for defining and implementing this National waste management effort. The Environmental Protection Agency (EPA) and the Nuclear Regulatory Commission (NRC) are responsible for providing a framework of criteria, standards, and regulations that will assure that the disposal methods developed by DOE are consistent with the achievement of the goal of safe, long-term waste isolation.

Within DOE, the responsibility for implementation of the National waste management program rests with the Office of Nuclear Waste Management. The nuclear waste management effort involves three major activities: the Defense Waste Management Program, the Spent Fuel Storage Program, and the Commercial Waste Management Program. The major programmatic effort in the Commercial Waste Management Program is the National Waste Terminal Storage (NWTS) Program.

The objective the NWTS Program is to develop the technology for the safe, environmentally acceptable isolation of high-level and transuranic nuclear wastes. Although alternative methods are being studied, major programmatic emphasis is on disposal of these wastes by emplacement in deep, stable, geologic formations. The NWTS effort includes comprehensive geological exploration and technology development programs.

The NWTS Program involves three separate but coordinated efforts: the Basalt Waste Isolation Project (BWIP) at Hanford, the Nevada Nuclear Waste Storage Investigations (NNWSI) at the Nevada Test Site (NTS), and the Office of Nuclear Waste Isolation (ONWI) Program. BWIP and NNWSI are examining the waste isolation potential of geologic formations and associated site-specific technical issues on their respective DOE reservations, while ONWI, which is based in Columbus, Ohio, is investigating other formations within the contiguous United States and is conducting a comprehensive generic technology development program.

This document is a compilation of papers presented at an NWTS Information Meeting, one of a series, planned to provide a comprehensive review of the progress of the ONWI Program, as well as an overview of the BWIP Program and NNWSI, for participants in the NWTS Program, interested members of the technical community, and the general public. The NWTS Program is open to peer and public scrutiny. The safe disposal of nuclear waste is a National problem and requires an informed National consensus for its resolution. The sponsorship of open information meetings exemplifies DOE's effort to develop this consensus.

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SESSION I
PROGRESS IN DISPOSAL OF NUCLEAR WASTE: AN OVERVIEW OF THE NWTS PROGRAM

IMPLEMENTING THE NATIONAL CONSENSUS ON DISPOSAL OF NUCLEAR WASTE

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Office of Nuclear Waste Isolation

In a democracy such as the United States, the term consensus is often utilized when referring to the governmental decision process. The consensus often is expressed in the form of majority or plurality vote. The consensus can be a political consensus such as a vote in various legislatures or it can be a social consensus as represented by the results of national or state elections or the vote of the electorate on certain initiative and referendum issues.

Many types of issues are debated in a democracy when seeking for the governmental consensus. The issues may be social issues such as national health care, or defense issues such as development of a new weapons system, or issues related directly to the development and regulation of new technology. We are discovering that new technology is a very important factor in the quality of life within the United States. However, the policies associated with developing technology have become very complex and it has been correspondingly difficult to develop a governmental consensus.

We have gone through an explosive development of high technology in the United States and in the world in recent years. Society has discovered that technology can provide significant benefits and that it also can have associated with it significant disadvantages such as environmental impacts, risks to human health and life, and a potential for misuse. The political process in this country has attempted to deal with the policies related to technology and it is increasingly apparent that this is a difficult process. In a democracy, the people and their elected representatives make the decisions regarding the development and control of technology. However, these same people frequently are not expert in the technology itself and, therefore, have difficulty in understanding and evaluating the full implications of that technology development. Therefore, a communication gap develops between the individuals charged with representing the consensus policy and those who are the "experts" in the technology.

One can observe this process by recalling examples of legislation developed in the last few years to control the negative aspects of technology development. For example, the Clean Air Act, Federal Water Pollution Control Act, Conservation and National Environmental Policy Act, Energy Reorganization Act of 1974, and many other examples are products of the growing national debate on how to exploit the advantages of technology and, at the same time, adequately protect society from its disadvantages. I maintain that our institutional experiment has not yet been developed to a successful conclusion; that is, we have not yet discovered an efficient method to communicate the attributes and disadvantages of technology to the decision makers in such a way that they can make their decisions in a fully informed and timely manner.

The safe long-term disposal of nuclear waste is another example of an issue resulting from high technology which requires an informed national consensus for its resolution. In fact, we could say that resolution of the waste disposal problem requires:

- A political consensus
- A technical consensus
- A social consensus.

The National Waste Terminal Storage program is organized and conceived to contribute to the development of the national consensus requisite for selecting and implementing disposal options for nuclear waste. The program is primarily technical and is, therefore, oriented toward developing the information and processes required for establishing a technical consensus on disposal options. However, the DOE recognizes that information developed within the program must be adequate to support the parallel development of national political and social consensus on the recommendations and solutions resulting from the technical work. The development of the national consensus on nuclear waste disposal is, in many ways, at the forefront of a real institutional experiment within this country to determine whether the democratic process can provide appropriate and needed solutions to a high-technology problem. In this Information Meeting, we will discuss the programs under way in the NWTS effort to develop the technical consensus and some of the preliminary results resulting from this work that will help to develop the requisite national consensus.

NUCLEAR WASTE TECHNOLOGY: ADDRESSING THE TECHNICAL ISSUES

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Early formulations of the U.S. nuclear waste-management strategies date back to the recommendations in the mid-50's by the National Academy of Sciences. At that time, NAS recommended nuclear waste disposal in underground salt beds as the preferred approach. These recommendations were generally confirmed by another NAS committee in the late 60's.

In the mid-70's, the nuclear waste program was extensively reexamined by ERDA. ERDA obligated itself to provide high-level repositories for the commercial industry. These repositories were to be licensed, and a generic environmental impact statement was to be prepared.

In 1978, at the direction of President Carter, an Interagency Review Group on Nuclear Waste Management was established. The final IRG report was issued in March, 1979. A key recommendation of this report was that, for interim planning purposes, the first disposal facilities should be limited to mined repositories. It was further recommended that nuclear waste disposal should proceed on a step-wise basis in a technically conservative manner, that a number of potential sites should be identified, and that a systems approach should be used to select the geologic environment, repository site, and waste form.

The Department of Energy has the responsibility for developing the technology required to manage the commercial radioactive wastes in a safe and environmentally acceptable manner. As part of this DOE mission, a "Draft Environmental Impact Statement" (DEIS) was issued for public comment in April of 1979.

The content and organization of the NWTS/ONWI technology program are intended to be responsive to the major technical uncertainties associated with the design bases for conventional geologic disposal. These uncertainties can be categorized in a variety of ways. A useful categorization into six basic areas of uncertainties is given in the DEIS, as follows:

- System data-base development
- Techniques for analyzing geologic formations
- Techniques for decommissioning and monitoring
- Thermal analysis and rock mechanics
- Rock-waste interaction
- Verification of safety and reliability.

Since repository design, construction, and waste emplacement are complex processes and since long-term performance of the repository is difficult to predict, a multibarrier, defense-in-depth design philosophy is being followed. Engineered, man-made barriers and geologic barriers are used to form a system of obstacles to potential radioactive nuclide migration.

The NWTS/ONWI technology program comprises more than 75 individual research contracts budgeted at about $30 million in FY 1979 and at a comparable amount for FY 1980. The program is organized into four subelements, as follows:

- The "Waste Package"
- The "Repository"
- The "Geology"
- The "System".

Program emphasis is placed on multibarrier waste-package development and design, the development of borehole-plugging and shaft-sealing techniques, in situ testing to verify near-field, near-term effects, and extensive performance modeling and earth science research efforts to ensure adequate far-field, long-term performance of the waste repository.

Considerable technical progress has been made during the past year and a well-formulated technology program is in place for FY 1980.

THE GEOLOGIC REPOSITORY—WHERE, HOW, WHEN?

M. Kehnemuyi
Office of Nuclear Waste Isolation

Where?

Several geologic formations are now being investigated for locating repositories for disposal of spent fuel assemblies or reprocessed high-level wastes from commercial nuclear power plants. The formations being investigated include rock salt, granite, shale, basalt, and tuffs.

Following a recommendation by the National Academy of Sciences in September, 1957, most of the scientific investigations in the United States had been centered on rock salt. Recently, activities have been initiated and accelerated to explore other media as possible host formations for repositories.

The studies include four rock salt formation regions, formations in two government reservations, and additional geohydrologic systems as follows:

(1) **Gulf Interior salt-dome formations in the states of Mississippi, Louisiana, and Texas.** Extensive field activities are now in progress in these three states. In Mississippi three domes are under investigation: the Richton and Cypress Creek domes in Perry County and the Lampton dome in Marion County. In Louisiana, the Rayburns and Vacherie domes in Bienville Parish are being investigated, and in Texas the Oakwood dome, which straddles the borders of Freestone and Leon Counties, and the Keechi dome in Anderson County are being studied. Deep exploratory drilling into the salt domes has been completed at Cypress Creek, Richton, Oakwood, Vacherie, and Rayburns. The cores are now being analyzed, together with data obtained from hydrologic explorations and geophysical surveys. Concurrently, environmental studies on wildlife habitats, land uses, surface hydrology, demography, and other nongeologic considerations are being conducted.

(2) **Paradox region anticlines and bedded formations in the state of Utah.** Field activities in this region have already started, and major drilling has also started. The field exploration activities are primarily centered in the areas of the Gibson dome, 25 miles south of Moab, and Elk Ridge, 20 miles west of Blanding, both within San Juan County. In 1978 in Salt Valley, which is located 25 miles northwest of Moab in Grand County, three deep-hole explorations ranging in depth from 1300 to 4000 feet were completed. In addition, the existing geologic and hydrologic data at Lisbon Valley are being examined for possible consideration of that area as a repository location.

(3) **Permian bedded formations in the states of Texas and New Mexico.** The Palo Duro basin in northwestern Texas is currently being studied for repository sites. The Waste Isolation Pilot Plant (WIPP) site is also located within this region in southeastern New Mexico.

(4) **Salina bedded salt formations extending from New York to Michigan.** These formations have been considered as possible locations for siting repositories. At present, there are no field activities, pending further negotiations with state authorities on the nature and timing of the studies to be performed.

(5) **The basalt formations in the Hanford Reservation in the state of Washington.** The objective of the Basalt Waste Isolation Program is to evaluate the use of the Hanford Reservation as a potential terminal storage site and to investigate the chemical and thermal suitability of basalt formations for waste disposal. Activities include evaluations to determine site suitability, and identify possible repository sites, tests to determine how nuclear waste will interact with basalt, and studies of the thermal and radiation effects. A Near Surface Test Facility is being constructed to carry out the tests.

(6) **Tuff, granite, and shale formations at the Nevada Test Site.** The Nevada Nuclear Waste Storage Investigations program is being conducted at DOE's Nevada Test Site near Las Vegas in southern Nevada. NNWSI is evaluating the location of a repository site at the NTS facility as well as its compatibility with continued weapons testing there. Unlike other areas in the NWTS program, the Nevada

Test Site provides a variety of geologic systems, meaning that tests can be conducted in granite, argillite (a compact clay rock cemented by silica), and tuff (a heat-fused volcanic ash). NNWSI's tasks include evaluating the area's use for a waste repository, identifying a specific site and the most suitable geologic media for a repository, determining the effects of weapons testing on available geologic media, and on-site testing of thermal effects of encapsulated spent nuclear fuel placed underground. Other studies evaluate the effectiveness of shale, granite, and tuff as a waste barrier and the impact, if any, of ground motion from weapons testing on repository design.

(7) **Studies initiated to identify regions of granite and argillaceous rocks where repositories may be sited.** These studies will be completed early in 1980, and the regions of interest will be designated.

(8) **A National survey on identifying suitable sites.** The objective is the identification of other sites that could provide containment for nuclear wastes on the basis of integrated evaluations of geologic, hydrologic, and environmental considerations.

How?

Site qualification and identification programs are based on a series of increasingly detailed studies to obtain geologic and environmental data. The scope of the studies narrows from regions to areas and then to specific locations.

The screening procedures take the following steps: (1) selection of a suitable geologic system for a waste repository; (2) determination of regions within such systems for further study; (3) recommendation of specific locations for further study; and (4) recommendation of preferred repository site(s).

When?

The complex undertaking represented by the siting, design, construction, and licensing of the first nuclear waste repository, and the timely development of the technology base needed to support these activities, can be approached only by mounting a very highly coordinated effort to achieve the sequential milestones established by a realistic schedule. All repository development programs utilize such schedules, and apart from the specified site selection date, they tend to be essentially identical in format and milestone intervals. The schedule for the NWTS/ONWI repository development program is depicted in Figure 1.

It can be seen that the critical path of the schedule tracks through milestones associated with site exploration, facility design, licensing, construction, and final operational checkout activities. Two time scales are shown on this schedule. The Strategy III scale of IRG ("Report to the President by the Interagency Review Group on Nuclear Waste Management", March, 1979) postpones the time of site selection, which is the most critical date in the repository development process, until a number of geologic formations, including closed hydrologic systems, have been adequately explored. These formations would involve such host media as salt, basalt, granite, tuff and shale. Under IRG Strategy II requirements, we would select a site on the basis of explorations of Gulf Interior Region salt domes and the basalt formation on the DOE Hanford Reservation in the state of Washington. It can be seen that the scheduled date of repository operational status differs by about 4 years under the two strategies.

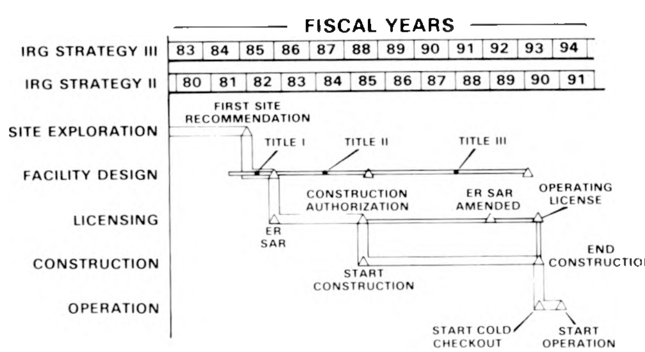

FIGURE 1. SCHEDULE FOR NWTS/ONWI REPOSITORY DEVELOPMENT PROGRAM

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STATUS REPORT ON STUDIES TO ASSESS THE FEASIBILITY OF STORING NUCLEAR WASTE IN COLUMBIA PLATEAU BASALTS

R. A. Deju
Basalt Waste Isolation Program

The Basalt Waste Isolation Program, operated by Rockwell International for the U.S. Department of Energy, is part of the National Waste Terminal Storage Program. The program, as presently structured, is aimed at assessing the feasibility and providing the technology needed to design and construct a repository for storage of commercial radioactive waste in the extensive basalts beneath the Hanford Reservation of DOE.

The Hanford site has been the location of nuclear energy work for several decades and is centrally located within the Pasco basin in the midst of the large basaltic flows (Figure 1).The previous and present commitments of this site to the nuclear field make it a logical place to be examined for potential use as a repository site.

The program is presently in the research and development phase to assess feasibility. Geologic site-selection studies are to be completed in September, 1981, to allow a feasibility decision at that time, whereupon if feasibility is proven and the DOE goes forward with the project, we would move into the project's licensing phase. Ultimately, it would be up to the Nuclear Regulatory Commission to grant a construction license and, later on, an operating license.

During the research and development phase, the program includes seven areas of study: geosciences, hydrology, multiple engineered barriers, test-facility design and construction, engineering testing, systems integration, and preliminary repository engineering design. In the next few paragraphs, I will attempt to detail the progress made to date in each of these areas.

Geosciences

The geosciences program is aimed at gathering the data required for selection and evaluation of potential repository sites in basalt. These studies lead to identifying repository target areas, and when a site is selected, they will thoroughly characterize the area to determine in great detail the extensiveness of individual basalt flows, the stability of the region, and the presence or absence of potentially hazardous geologic structures.

Studies to date have included reconnaissance regional studies and local studies of a more intensive nature within the Pasco basin, where the Hanford site is located. As part of the regional studies, a survey of published and unpublished documents concerning the geology of the Columbia plateau was completed. In addition, mapping of the basalt within the Columbia plateau and the overlying Late Cenozoic sediments has been conducted and is essentially completed. The stratigraphy of Columbia river basalt group has been revised and its understanding considerably expanded.

The Pasco basin studies have included a definition of the local stratigraphy of the basalts and an assessment of the viability of using the chemical and magnetic properties of the basalts for stratigraphic definition. Extensive mapping has been conducted and continues to be conducted in structurally significant areas of the Pasco basin. Drilled holes within the Hanford site and vicinity are being used to understand the structural relations of individual basalt flows. The geologic mapping in the field is also aided by geophysical studies, including seismic, aeromagnetic, magnetotelluric, and gravity prospecting.

Prior to site selection, all of the geological information will be incorporated into a comprehensive report, which will be subjected to extensive peer review.

Hydrology

The hydrology program parallels the geologic studies. The program has emphasized the gathering of data to characterize the groundwater systems underlying the Pasco basin and the modeling of such data to evaluate radiocontaminant migration to the biosphere. Studies to date indicate that the basalts in the Pasco basin are extremely tight, with the fractures in the deeper basalts being generally clay-filled.

The hydrology program has heavily depended on the existence of test holes at Hanford and in its vicinity for obtaining all needed property measurements. Holes tested range in depth from 3000 to over 10,000 feet (Figure 2). Additional holes are being planned for the very near future.

These holes form the basis of information being used to model the hydrology of the Pasco basin. Rockwell Hanford Operations interpreted drill stem tests from well RSH-1 southwest of Hanford. Our analysis shows hydraulic-conductivity values between 10^{-7} and 10^{-9} cm/sec within the dense basalt flows themselves. Science Applications, Inc., personnel conducted more sensitive tests and found hydraulic conductivities ranging from 10^{-7} to 10^{-13} cm/sec in the zones tested. At present, they are conducting additional tests. In parallel with the field program, we are conducting model adaptation and sensitivity analysis of our computer model for the hydrology of the basin.

Engineered Barriers

The emplacement of nuclear waste in a geologic repository may cause physiochemical perturbations to the surrounding environment. The multiple engineered barriers program attempts to identify, from a physicochemical standpoint, the features of various barriers to the transport of radioactive contaminants. The program is looking at four potential barriers: the waste form, the container, the overpack, and the rock. In addition, a borehole-plugging system is being developed as a final barrier once the repository is sealed and abandoned.

The multiple barriers program includes (Figure 3) a material-characterization task where all the components of the multiple-barrier system are analyzed, and a reaction study where radionuclide sorption tests and hydrothermal tests are being conducted to provide the source terms required for radionuclide-transport calculations. Decision-analysis techniques are being used to aid in optimizing the selection of a multiple-barriers system.

Near-Surface Test Facility

During the early phase of the Basalt Waste Isolation Program, the need for in situ thermal and mechancial testing of basalt was identified. This need for engineering data, to qualify basalt as a repository medium and to provide the basis for repository design, could be met by construction of a near-surface multipurpose facility for in situ testing of basalt. Further, the facility would serve for the demonstration of placement, storage, and retrieval of nuclear waste canisters in an underground basalt environment and the demonstration of the effectiveness of waste-monitoring systems. Detailed planning on this facility was started in October, 1977. Construction of the Facility began in June, 1978, and the first test is scheduled for a mid-1980 start-up. Work is presently on schedule.

The Near-Surface Test Facility is located approximately 340 feet below the land surface, and approximately 500 feet into the Pomona basalt. This allows a sufficient portion of basalt to remain undisturbed below the test room for the conduct of the test. The facility consists of three entrance tunnels and two test areas, one for heater simulation and one for spent-fuel tests (Figure 4). Results from the heater tests will provide data on borehole decrepitation, thermal stability, structural integrity, temperature, and displacement fields, and the influence of fractures and joints upon in situ basalt properties. These studies will provide the basis for the design of key repository elements such as canister storage, borehole criteria, borehole-liner performance, acceptable waste-canister power levels, storage borehole-array criteria, and repository step-loading evaluation. In addition, the in situ testing program should aid in the verification of mathematical models of the thermomechanical response of a repository. The spent-fuel testing will serve to study the combined effect of radiation and thermal response due to decay. In addition, a combined heater-spent fuel array will be used to heat a large room and examine scale effects and room-stability considerations.

Tunneling work is now complete. The heater test area required the excavation of approximately 21,000 cubic yards of material from the two portal areas and the development of approximately 1800 feet of underground workings. Tunnels in the heater test area vary in size from 8 feet in diameter for the east access tunnel to 23 feet in diameter for the accelerated heater test room and the extensometer room.

The nuclear waste test area required the excavation of an additional 25,000 cubic yards of material from the portal area and another 1200 feet of underground excavation. The test room in this area is 28 feet high. All test areas have unsupported roofs. A flash coat of shotcrete, however, has been added to the roof for rock stability.

Engineering Testing

The engineering testing program includes the determination of engineering properties of basalt needed for conceptual engineering-design studies and qualification of basalt as a medium for storage of nuclear waste. The program began with an extensive literature search of the engineering properties of basaltic rocks. After this literature search was completed and existing data were tabulated, samples from numerous core holes corresponding to many depth horizons from wells within the Hanford Site were subjected to extensive thermal and mechanical tests. In addition, the test program has been designed to gain basic input data from numerical models, as well as to determine lateral and vertical variations of properties between and within individual basalt flows.

Systems Integration

The systems integration program, as the name implies, is responsible for the use of systems- and decision-analysis techniques in the integration of the research and development effort. Primary areas of concern involve repository siting: What constitutes the feasibility of building a basalt repository? When is a site qualified? And what are the licensing concerns and requirements? Work in all areas is under way.

Repository Engineering

One final area of involvement has been the development of the preconceptual design of a basalt repository. The preconceptual design, recently completed, includes five areas: surface facilities, access and isolation shafts, subsurface facilities, waste handling systems, and surface and subsurface service systems.

The preconceptual design also covers the final stage of operations, including the ultimate plugging and abandonment of the facility.

The completed preconceptual design will serve to produce a first-order cost estimate for the repository, schedules for construction, and a preliminary safety assessment.

Conclusion

At this time we have made great progress toward establishing the feasibility of using the deep basalts beneath the Hanford Site for underground disposal of commercial nuclear waste. However, some complex issues such as the nature of the deep hydrology and the in situ thermal response of the basalts remain to be fully assessed. These and other related issues are the subject of present and future studies.

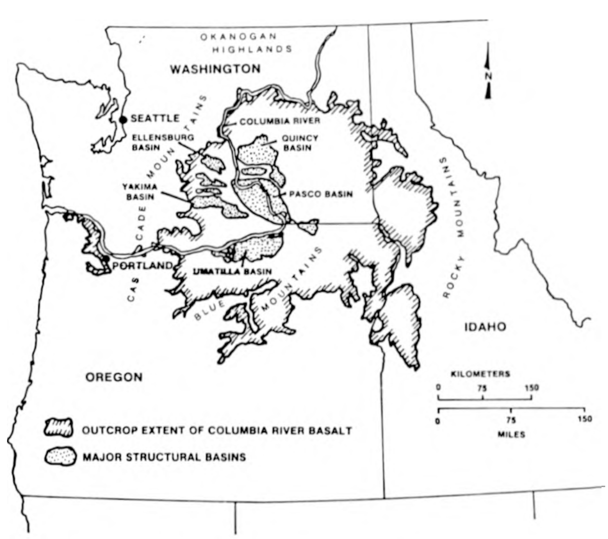

FIGURE 1. THE COLUMBIA PLATEAU

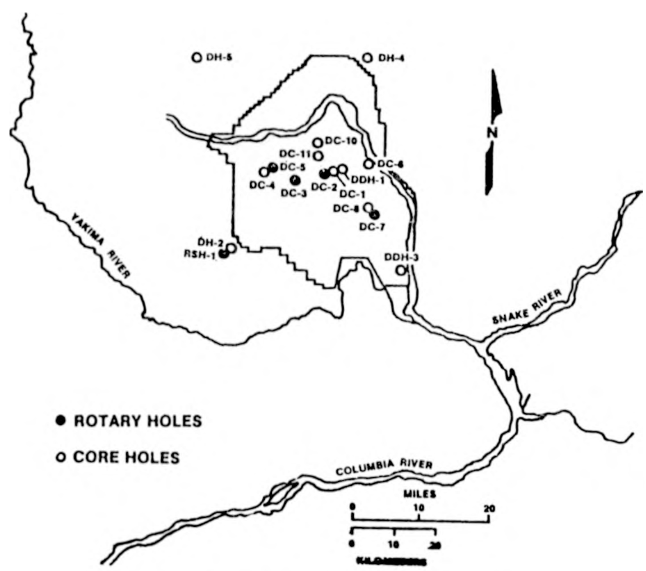

FIGURE 2. DRILLING LOCATIONS

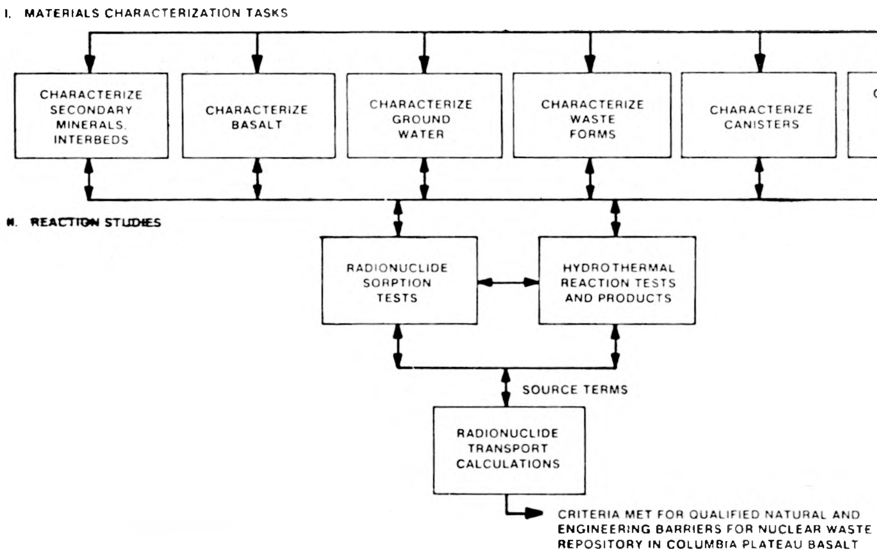

FIGURE 3. SIMPLIFIED DESCRIPTION OF ENGINEERED-BARRIER ACTIVITIES

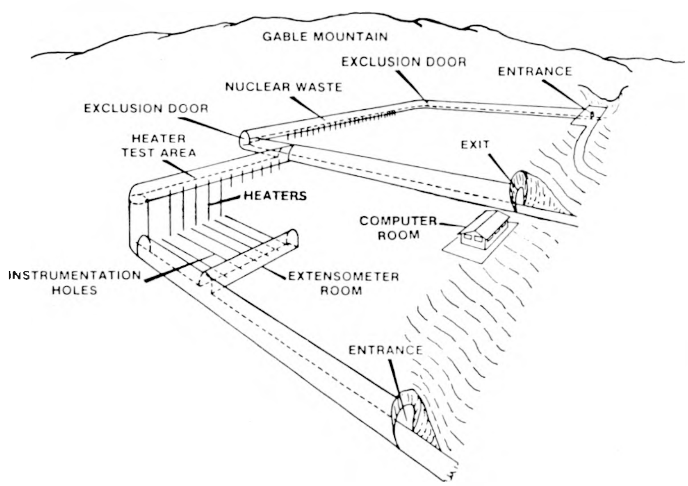

FIGURE 4. NEAR-SURFACE TEST FACILITY

THE NEVADA NUCLEAR WASTE STORAGE INVESTIGATIONS

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Sandia Laboratories

The Nevada Nuclear Waste Storage Investigations are part of the National Waste Terminal Storage Program and have as their objectives: (1) evaluation of the Nevada Test Site (NTS) and nearby areas for their potential as sites for the permanent isolation of commercial spent fuel or high-level radioactive wastes; and (2) research and development in support of the NWTS Program. The project is managed by the Department of Energy's Nevada Operations Office, with technical participation by Lawrence Livermore Laboratory, Los Alamos Scientific Laboratory, Sandia Laboratories, the United States Geologic Survey, and various NTS contractors. The NTS, which is on Federally owned, restricted-access land in southern Nye County, Nevada, has been primarily dedicated to nuclear weapons testing. The two initial issues addressed by the site-evaluation investigations are the compatibility of geologic isolation of radioactive wastes with the prime NTS mission of nuclear weapons testing and the demonstration that a suitable geohydrologic setting with an acceptable disposal medium exists within the NTS. The southwest portion had been identified as one area of the NTS where a repository could be sited on the basis of noninterference with weapons testing. FY 1979 efforts directed at potential repository site evaluations include a wide variety of geologic, geophysical, and media studies which are focused on areas within the southwest area of NTS. In addition, spent-fuel interim storage tests in support of the NWTS Program were initiated during FY 1979.

The geological setting of the NTS [1] and adjacent areas is highly complex, providing a wide variety of potential emplacement media and structural-hydrologic environments. This area of the Basin and Range Province was subjected to at least two major episodes of mountain building. The first, about 150 million years ago, was characterized by compressional stresses, thrust faulting, folding, and granitic intrusions which deformed and slightly metamorphosed a 30,000$^+$-foot-thick sequence of sandstones, shales, and limestones. The second began approximately 30 million years ago with the eruption and deposition of predominantly silicic volcanic rocks. Tensional stresses and resultant normal faulting accompanied the volcanism and produced the present basin-range topography. Silicic volcanism was followed by minor basaltic eruptions about 6 to 9 million years ago. Basaltic eruptions and basin-range faulting are still active in parts of the present Great Basin.

Groundwater basins in the NTS region are closed flow systems and commonly encompass several topographic basins. The water table varies from 2000 to 3000 ft or more below the surface in the ranges and asymptotically approaches the surface toward discharge areas. Discharge occurs in discrete topographically low areas characterized by numerous springs and seeps. For the NTS groundwater basin, partial discharge occurs approximately 20 miles south of the NTS and the remainder in and adjacent to Death Valley, California, about 50 miles south of the NTS.

FY 1979 geologic studies in support of potential site evaluation have included field and aerial reconnaissance, geologic mapping, and dating of faults and volcanic events. Nonpenetration geophysical techniques—including electrical, gravity, seismic, and aeromagnetic surveys—have been employed to characterize the subsurface geology in four areas of the southwest portion of the NTS. Based on these geologic studies, a programmatic decision was made during FY 1979 to concentrate geotechnical data gathering on a portion of Yucca mountain. During FY 1980, a key stratigraphic hole will be drilled at Yucca mountain.

Hydrologic studies include the development of a regional hydrologic model and a supporting paleohydrology study. The regional hydrologic model will be used to characterize groundwater pathways between candidate repository sites and points of present or possible future discharge. Results from this model will be coupled with transport equations to predict rates and directions of movement of radioactive species if they should be released from a repository. The objective of the paleohydrology study is to define water-table depth, gradients, and pathways to points of groundwater discharge during pluvial cycles in southern Nevada.

Because of the numerous rock types that occur throughout the NTS and adjacent areas, the Nevada project is able to consider multiple options for selection of an emplacement medium. Characterization of four media which occur in the southwest NTS, argillite, granite, tuff, and alluvium, has included field and laboratory experiments and modeling studies. Petrologic and sorptive studies of some of these media [2-5] have been initiated. During FY 1979, a field heater test in argillite [6,7] was completed. While data analysis and associated modeling will continue into FY 1980, early indications are that the dehydration of expandable clays resulted in the opening of preexisting joints near the heater.[6,7] This result indicates that, depending on the impact of this near-canister fracture permeability increase, thermal or mineral constraints might need to be placed on a high-level waste repository using argillite as an emplacement medium. The results of initial seeping studies of tuffs were presented to the National Academy of Sciences, Committee on Radioactive Waste Management, late in FY 1978.[8] This presentation identified the favorable, unfavorable, and unresolved issues for tuffs. FY 1979 work related to the evaluation of tuff as an emplacement medium focused on the unresolved issues, including preparations for a FY 1980 field experiment to evaluate the effect of heat on the water content of welded tuff.

One example of NWTS program support is a test to simulate—with 11 spent fuel canisters, 6 electrical simulators and supplementary heaters—an accelerated thermal history of a unit cell of a conceptual spent-fuel respository in granite. This test[9], known as the Climax Spent Fuel Test, will be carried out over a period of 3 to 5 years at a depth of 1400 ft in the Climax Stock granite. The effects of both heat and radiation on the granite will be compared with the effects of heat alone, and both thermal-response and rock-mechanics data will be collected and compared with model predictions. This test, in support of the NWTS Program, will start during FY 1980. The test will utilize the current capabilities of the spent fuel handling and packaging program facilities at NTS to encapsulate the spent fuel assemblies.

The spent-fuel encapsulation and interim storage technology development and testing work will be described in detail in a paper titled "EMAD Support for NWTS Experiments and Demonstrations" to be presented at this meeting in Session IIC.

The diversity of rock types and structural settings at the NWTS requires a broad waste-management perspective of site and media options. This project has developed a research program that addresses its unique range of site-selection options as well as supports the national effort to find a satisfactory means to dispose of nuclear waste.

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SESSION IIA
TECHNICAL STUDIES IN THE NWTS/ONWI SCIENCE AND TECHNOLOGY PROGRAM

EARTH SCIENCES

THE USE OF STABLE DECAY PRODUCTS FOR DATING GROUNDWATER

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The age of groundwater (the time since contact with the atmosphere) can be determined by measuring the accumulation of stable products from radioactive decay of elements naturally found in rocks. The age of groundwater is a significant factor in the evaluation of safety of proposed subsurface repositories for nuclear waste. Although the groundwater age cannot be used alone to evaluate the safety of a repository, knowledge about its age contributes greatly to the understanding of the hydrologic system which must be used in such an evaluation.

Stable-Decay-Product Dating Method

Radioactive uranium, thorium, and potassium are present in nearly all rocks. The decay products are generated continuously within the rock, some of them enter the water. If the generation rate of the decay product is known and parent radionuclide content of the rock is known, then a minimum residence time for the water to be in contact with this type of rock can be calculated from the amount of decay product present in the water.

The principal assumption involved in this calculation is that the transfer rate of the decay product from the rock to the water is the same as the generation rate. For a hydrologic system subject to changing conditions such as initial saturation or a radical change in salinity, this assumption may not be valid. However, if the hydrogeological history of the region shows that the groundwater system has not undergone significant change, the release rate of the decay products from the rock to the water may have reached a steady state.

Decay products that are also noble gases are particularly useful for determining the age of groundwater because they do not react with the underground fluid/rock system. Such decay products are helium from uranium and thorium, argon from potassium, neon from an alpha-neutron reaction with oxygen-18 and fluorine-19, xenon from uranium and thorium, and krypton from uranium and thorium.

Of these elements, helium from uranium and thorium is the most useful because it is produced in adequate amounts to determine ages greater than 10,000 years. The others are produced in such small quantities that even at ages of 100 million years, measurement would be difficult.

The helium content of groundwater was used in determining an age of 840,000 years for water in the crystalline metamorphic rocks underlying the Savannah River Plant near Aiken, South Carolina.

Comparison with Other Dating Methods

Several radioisotopes are produced continually in the atmosphere (for example, tritium and carbon-14), and these have a constant ratio to the nonradioisotopes of the same elements (hydrogen and carbon-12). When precipitation becomes groundwater and is thus removed from contact with the atmosphere, this ratio changes as the radioisotope decays. From the known rate of decay and this ratio, groundwater age can be calculated. The age of groundwater that can be determined by this method is limited because the radioactive isotope eventually decays below the detectable limit. Currently, the use of tritium is limited to $\sim$ 40 years and carbon-14 to $\sim$ 50,000 years. These limitations may be extended by the development of more sensitive measuring equipment or by using isotopes with longer half-lives (chlorine-36 and krypton-81). Some of these atmospheric isotopes (carbon, for example) are chemically active and may be removed or added to the groundwater by chemical reactions with the rock. Thus, hydrogeochemistry must be taken into account. By contrast, decay products from radioisotopes in the rock become greater in concentration with age and are commonly noble gases that are not chemically active.

The disequilibrium between the amount of uranium-234 and its parent uranium-238 has been used as an indication of age, but this approach is dependent on a thorough understanding of the radiochemistry of uranium. Changes in the structure of molecules dissolved in water, such as amino acids, may also be useful, but much research remains to be done on this method.

Current Program

The Savannah River Laboratory is investigating the measurement of accumulated decay products in determining groundwater age. The program consists of determining the change in helium concentration in the groundwater along a flow path.

HIGH-FREQUENCY ELECTROMAGNETIC BOREHOLE METHODS

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Inhomogeneities and structural flaws in a rock mass indicate possible problems for its use as a repository. Use of radiowaves seems to offer a good means of finding such inhomogeneities and flaws. Unfortunately, wave probing of rock at present is comparable to seismic prospecting in the 1940's and 1950's: some basic capabilities exist, but considerable improvement is needed in (1) instrumentation, (2) understanding the wave-propagation phenomena, and (3) interpretative tools and techniques.

Electrical properties of rocks are very strongly affected by the water they contain. Tight-pored, unfractured rocks contain very little water and are therefore electrically resistive, whereas more fractured or porous rocks containing more water are less resistive. Thus, a formation's electrical resistivity in a gross way is proportional to its suitability as a repository medium. The more resistive a rock is, the better electromagnetic (radio) waves travel through it, the more likely it is to be suitable for radioactive waste storage.

Rock salt is the medium best suited to radio probing, largely due to its dryness. Stewart and Unterberger[1] reported propagation through more than 1200 m of salt, working in a dry salt mine. Unterberger (personal communication, 1979) reported significantly reduced penetrations when working in wet salt mines. He attributed this to loss in the wet salt, but an equally valid hypothesis is that back-scattered energy from the transmitted beam is sufficient to mask returns even from perfect reflectors.[2] This is an example of our lack of understanding of radiowave propagation in geologic media. We cannot adequately select a radar operating frequency until we have a characterization of the radiowave propagation, absorption, and scattering in the salt. These measurements involve bulk variations in the salt, and, therefore, cannot be done in the laboratory. A significant research effort will be needed to determine these rock characteristics.

To be of practical use in repository exploration, radar must be functional from boreholes; mined openings will not be available for radioprobing activities until construction has started. Radar also will be used in the excavation stage of the program. Use of boreholes introduces many practical problems for radar work:

(1) When a borehole is wet-drilled, what are the effects of the borehole fluids on antenna performance? Do the holes need to be pumped dry to conduct meaningful radar experiments?

(2) What are the interactions between the logging cable and the antennas? Is it necessary to use fiber-optic links to isolate the antennas from the cables?

(3) Is it feasible to achieve azimuthal directionality in the antenna? Since directional antenna positioning cannot be provided, can rotation and directional monitoring be used?

We have started to answer some of the questions about borehole fluid effects by simulating a brine-filled borehole on an antenna range. For a borehole diameter less than 3 percent of the wave length in the surrounding medium, we found that centering the antenna in the hole is not critical. There was only a 20 percent variation of antenna pattern when the antenna was pushed against a wall of the simulated borehole. There was, however, a loss of signal by a factor of 30 (30dB) per antenna, or a factor of 1000 in the full transmit/receive process. We are experimenting with ways to ameliorate this signal loss without pumping the hole dry.

Radar methods can be expected to be useful tools during the exploration, excavation, and monitoring phases of repository activity. The method is in its infancy, however, and a substantial amount of basic work is needed before operational status can be claimed. That work is well under way.

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LABORATORY INVESTIGATIONS ON THE HYDRAULIC AND THERMOMECHANICAL PROPERTIES OF FRACTURED CRYSTALLINE ROCKS

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J. E. Gale
University of Waterloo

Introduction

The safe underground storage of radioactive waste requires an understanding of the behavior of crystalline rocks under the influence of thermal and hydraulic stresses. The success of a repository will depend upon the use of such information in design and development considerations. Fracture systems in such rocks play a dominant role in the thermomechanical, as well as the hydraulic, behavior. Lawrence Berkeley Laboratory (LBL) is therefore conducting laboratory studies to investigate the laws of fluid flow in fractures, the dependence of fracture conductivity on the size of the sample tested, and the thermomechanical properties of both intact and discontinuous samples of crystalline rocks. This work provides support to the NWTS/ONWI Geologic Repository Program and is part of the Fractured Rock Studies Project and the Swedish-American Cooperative Program on Radioactive Waste Storage in Mined Caverns in Crystalline Rock[1] of LBL.

Stress Flow Behavior and the Cubic Law for Fluid Flow in a Fracture

Models of the movement of fluids in a fracture are generally based on the analogy of flow between parallel plates. For steady, laminar, isothermal flow, the flux per unit head can be expressed as:

$$Q/\triangle h = \frac{C}{f}(2b)^3$$

where Q is flow rate; $\triangle h$ is difference in hydraulic head, C is a constant depending upon flow geometry and fluid properties, f is a factor accounting for roughness and/or nonparallelism, and 2b is the fracture aperture. Equation (1) expresses the "cubic law" for fracture flow. Its validity has been established for laminar flow through open fractures (i.e., fractures where the surfaces are not in contact)[2,3]. It is important to establish whether or not the cubic law holds for closed fractures in which the surfaces have some degree of contact and the effective aperture depends upon the stresses acting across the discontinuity. To address this question, LBL, the University of California, Berkeley, and the University of Waterloo have performed tests on fractures artificially induced in intact samples of basalt, granite, marble, and other rocks. Radial and straight flow of water through fractures subjected to normal stress at ambient conditions has been studied[3-5]. Data from straight flow in fractures in granite are typical and will be used here to summarize the results.

Figure 1 shows the geometry of a 0.121-m-wide by 0.207-m-long by 0.155-m-high sample. A straight flow field was obtained by closing the sides, keeping the ends open, and injecting water through a small hole into a groove across the middle of the fracture plane[4]. LVDT's mounted across the fracture measured normal displacements. Normal stresses were applied in cycles of increasing and decreasing load. Figure 1 shows the relationship between stress and permeability. $Q/\Delta h$ decreases with increasing load and repeated loading cycles further reduce the flow rate, but the fracture does not completely close.

To check the validity of Equation (1) it is necessary to know the true fracture aperture, 2b. Equation (1) can be restated:

$$Q/\Delta h = \frac{C}{f}(2b_a + 2b_r)^n$$

where: $2b_a$ is the apparent aperture after a given stress history and is determined from the net fracture deformation, ΔV, and the maximum fracture deformation, ΔV_m, due to application of large stress; and $2b_r$ is the residual aperture after application of large stress (see Figure 2). If f is assumed unity, then the unknowns n and $2b_r$ can be determined[6]. Table 1 shows the results of the analyses (fitted values of n and $2b_r$) and includes values of $2b_r$ calculated by assuming that Equation (1) is valid. The separately calculated values of $2b_r$ agree closely and $n \cong 3$, which demonstrates the validity of the cubic law.

By adopting n = 3, values of f and $2b_r$ can also be determined from Equation (2).[6] Table 2 shows that values of $2b_r$ obtained in this way agree even more closely with values calculated from Equation (1). The computed values of f are reasonable and all greater than unity, which agrees with previous research results.[4,7]

Figure 3 shows experimental results expressed in the form of Equation (1) and in the alternate form relating friction factor, ψ, to Reynolds number, Re.[4,7] Good agreement with the cubic law is shown over a wide range of flow rates. It is concluded that, in simple fractures unaffected by weathering or shear movement, the cubic law is valid without regard to rock type for closed as well as open fractures. Permeability is uniquely defined by fracture aperture, regardless of stress history. The relationship between roughness and/or non-parallelism is linear and accounted for by the factor f in Equation (1).

Effect of Sample Size

The fracture-permeability studies included radial flow tests on 0.95-m-diameter by 1.9-m-high cores.[3,5] These tests used LBL's large triaxial testing facility (Figure 4). The machine has 5.2-MPa confining-pressure and 17.8-MN axial-loading capacities. Figure 5 compares stress-flow relationships obtained from these tests with results from smaller samples[4] and from in situ tests on 1-m^2 fractures[8]. At high normal stresses a limiting conductivity is reached, indicating that a limiting fracture aperture is approached[9]. This limiting fracture conductivity increases with increasing fracture area. This suggests that measured fracture conductivity is sample-size dependent and that tests on small specimens may yield unconservative underestimates of permeability.

For a systematic investigation of sample-size effects, LBL and the University of Waterloo are gathering specimens of naturally fractured rock ranging in diameter from 10 to 92 cm. Figure 6 shows a massive block of granite at the Charcoal Black quarry, Cold Spring, Minnesota, from which a 0.91-m-diameter by 1.83-m-high sample containing a fracture at mid-height will be cut.

Ultralarge Stripa Core

A 0.94-m-diameter by 1.78-m-high granite core has been obtained from the Stripa mine in Sweden. The 3629-kg core has been capped with reinforced concrete and the fracture geometry mapped over the surfaces and from a 7.62-cm-diameter hole drilled through the long axis (see Figures 7 and 8). The fractures are 1 mm or less wide and filled with chlorite, calcite, and altered muscovite mineralization. Fracturing is pervasive, but, based on preliminary falling head packer tests, the flow paths are dominated by the major intersecting horizontal (A through C) and steeply inclined (D through F) fracture sets. Permeability testing of this core under axial load in the large triaxial machine will assist in understanding the hydrology of the Stripa mine and allow comparison of simple radial-flow model results with data from a complex system of natural fractures.

Thermomechanical Rock Properties

Field investigations at Stripa[10, 11] have produced a number of interesting results which cannot readily be explained by thermoelastic theory. For example, thermal expansion of the granite adjacent to the emplaced heater canisters is measured to be less than half that predicted for intact rock. Furthermore, it is widely believed that the strength of rock is size dependent, and limited experimental data support the view that the strength[12] and hydraulic conductivity of discontinuities[9] depend strongly on size, at least in the range from centimeters to about a meter.

An attractive method of endeavoring to resolve and understand these differences involves laboratory testing of the thermomechanical properties of rock specimens of intact and discontinuous rock. LBL is pursuing this problem in two related efforts. A small-scale testing machine capable of handling samples from 0.05 to 0.15 m (2-6 in.) in diameter is nearing completion and will be operational in FY 1980. Large quantities of oriented core have been collected in Stripa and are now ready for testing at Berkeley. The measurements will include: thermal expansion, thermal conductivity, Young's modulus and Poisson's ratio at confining pressures up to 140 MPa and axial loads up to 2.7 MN. The system can operate at temperatures up to 200 C.

The second effort involves a much larger testing facility that is only in the planning stages. We have a preliminary design for a triaxial cell capable of handling rock specimens up to 0.76 m (30 in.) in diameter and 2.28 m (90 in.) in height. The cell is being designed to operate at temperatures up to 300 C and confining pressures up to 70 MPa and will be built to operate in the load frame of the machine shown in Figure 4.

This new test cell will enable the size-effect phenomena for both mechanical and thermomechanical loading to be studied in detail on intact and fractured rock specimens. It should be noted that although such large-scale tests are orders of magnitude more expensive than typical laboratory tests on specimens measuring centimeters in diameter, they are also orders of magnitude less expensive, quicker, and more certain than the only other alternative, namely, heavily instrumented field experiments. The principal scientific advantage of large scale laboratory tests over field tests is that the specimen being tested and the conditions of stress, strain, temperature, and pore fluid pressure and chemistry can be well controlled and defined over wide ranges of these conditions, which is never the case in field tests.

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TABLE 1. RESULTS OF LEAST SQUARES FIT FOR PARAMETERS OF n AND $2b_r$

Sample	Run	Fitted n	Residual Aperture ($2b_r$) μm Fitted	Residual Aperture ($2b_r$) μm Calculated[a]
Granite[b]	1	3.04	9.0	7.9
	2	3.03	6.7	6.7
	3	3.01	11.6	11.4
Granite[c]	1	3.07	5.1	4.4
	2	3.04	4.0	3.2
	3	3.06	13.1	10.9
Basalt[c]	1	3.08	10.5	10.0
	2	3.10	10.8	10.4
	3	3.05	9.9	9.8
Marble[c]	1	3.06	2.5	4.0
	2	3.06	2.2	4.0
	3	3.01	18.2	18.1

(a) Calculated from Equation (1).
(b) With straight flow.
(c) With radial flow.

TABLE 2. RESULTS OF LEAST SQUARES FIT FOR PARAMETERS OF f AND $2b_r$

Sample	Run	Fitted f	Residual Aperture ($2b_r$) μm Fitted	Calculated[a]
Granite[b]	1	1.21	8.8	7.9
	2	1.15	6.6	6.7
	3	1.04	11.6	11.4
Granite[c]	1	1.49	4.8	4.4
	2	1.29	3.8	3.2
	3	1.32	12.4	10.9
Basalt[c]	1	1.45	9.8	10.0
	2	1.65	9.9	10.4
	3	1.28	9.6	9.8
Marble[c]	1	1.36	2.2	4.0
	2	1.36	1.8	4.0
	3	1.05	18.2	18.1

(a) Calculated from Equation (1).
(b) With straight flow.
(c) With radial flow.

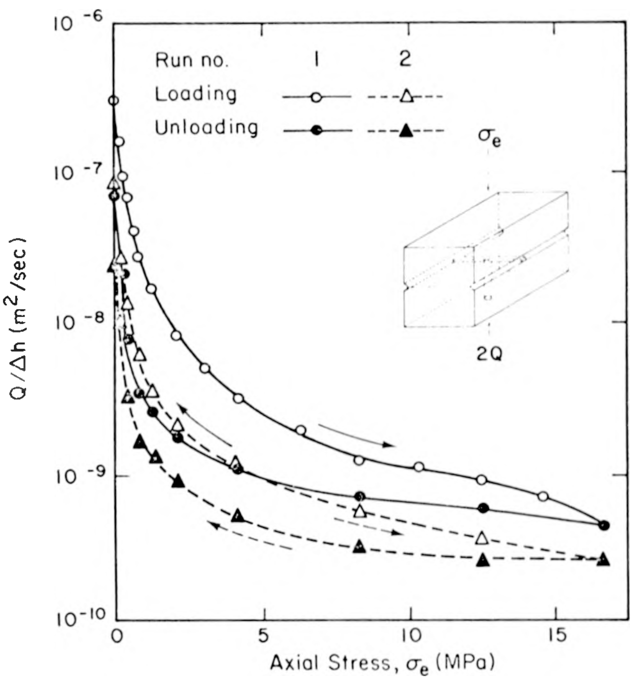

FIGURE 1. STRESS-FLOW RELATIONSHIPS; STRAIGHT FLOW IN GRANITE

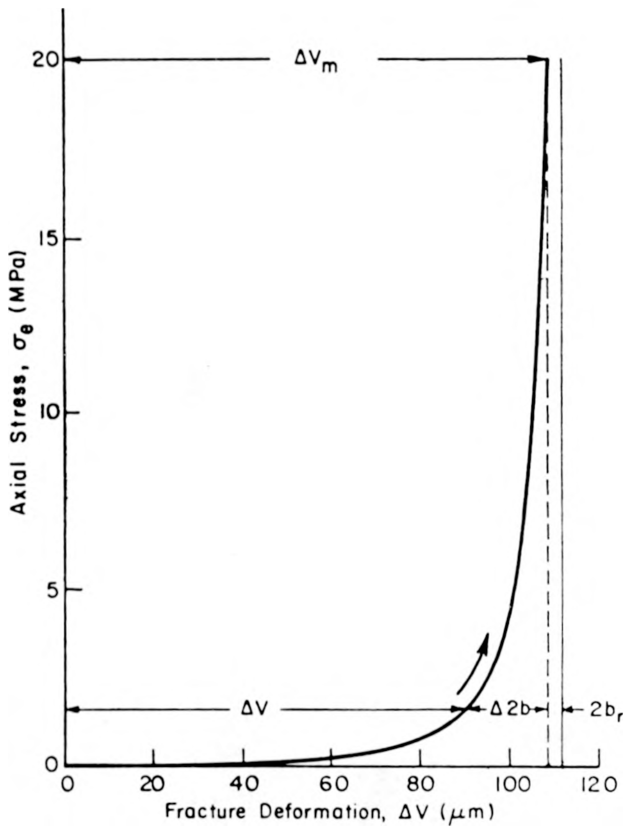

FIGURE 2. STRESS-DEFORMATION BEHAVIOR OF FRACTURE

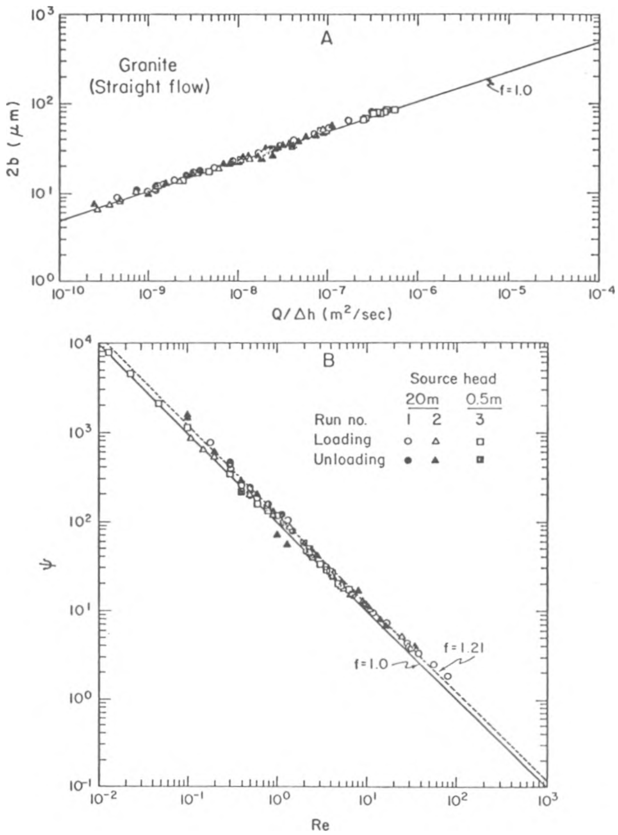

FIGURE 3. COMPARISON OF EXPERIMENTAL RESULTS FOR STRAIGHT FLOW THROUGH TENSION FRACTURE IN GRANITE WITH CUBIC LAW

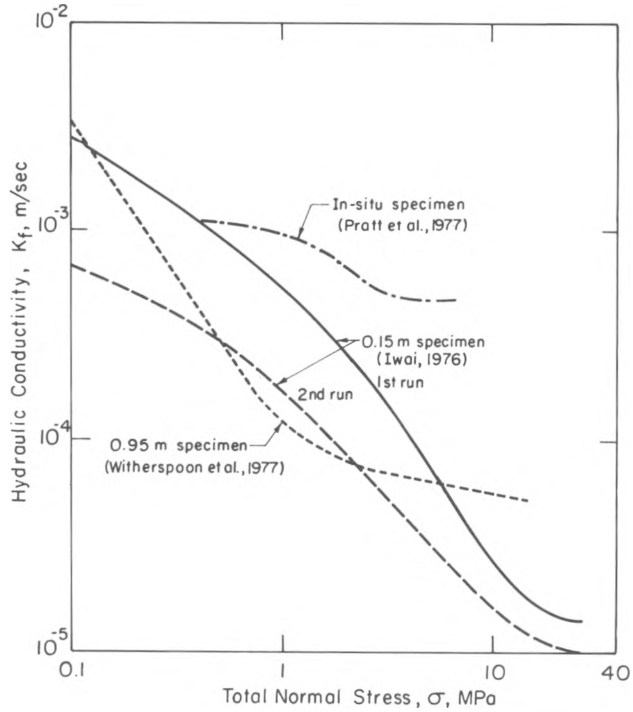

FIGURE 5. STRESS-FLOW RELATIONSHIPS FROM VARIOUS SIZE SPECIMENS

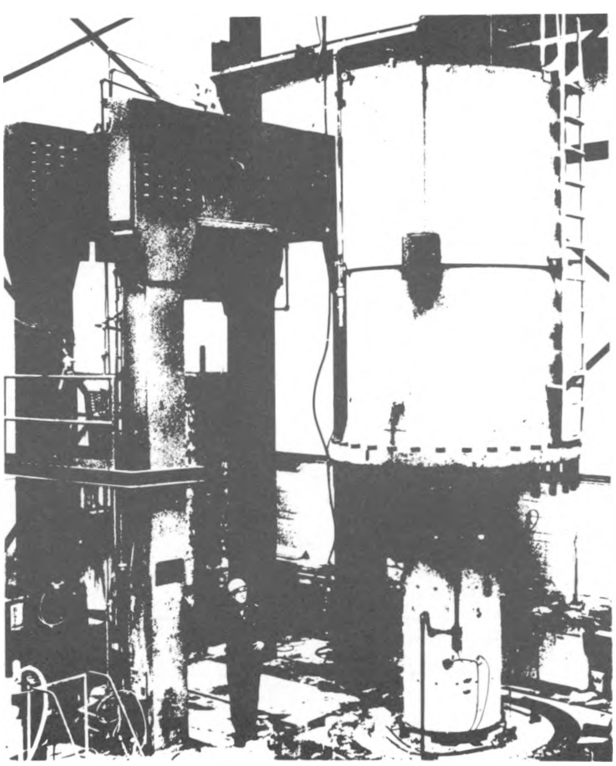

FIGURE 4. LARGE TRIAXIAL MACHINE

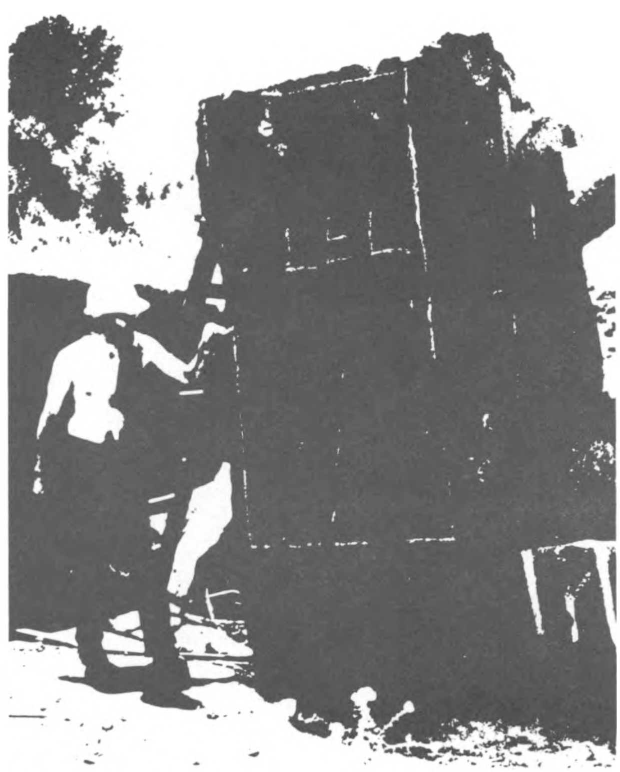

FIGURE 6. FRACTURED BLOCK OF CHARCOAL BLACK GRANITE

FIGURE 7. ULTRALARGE STRIPA CORE

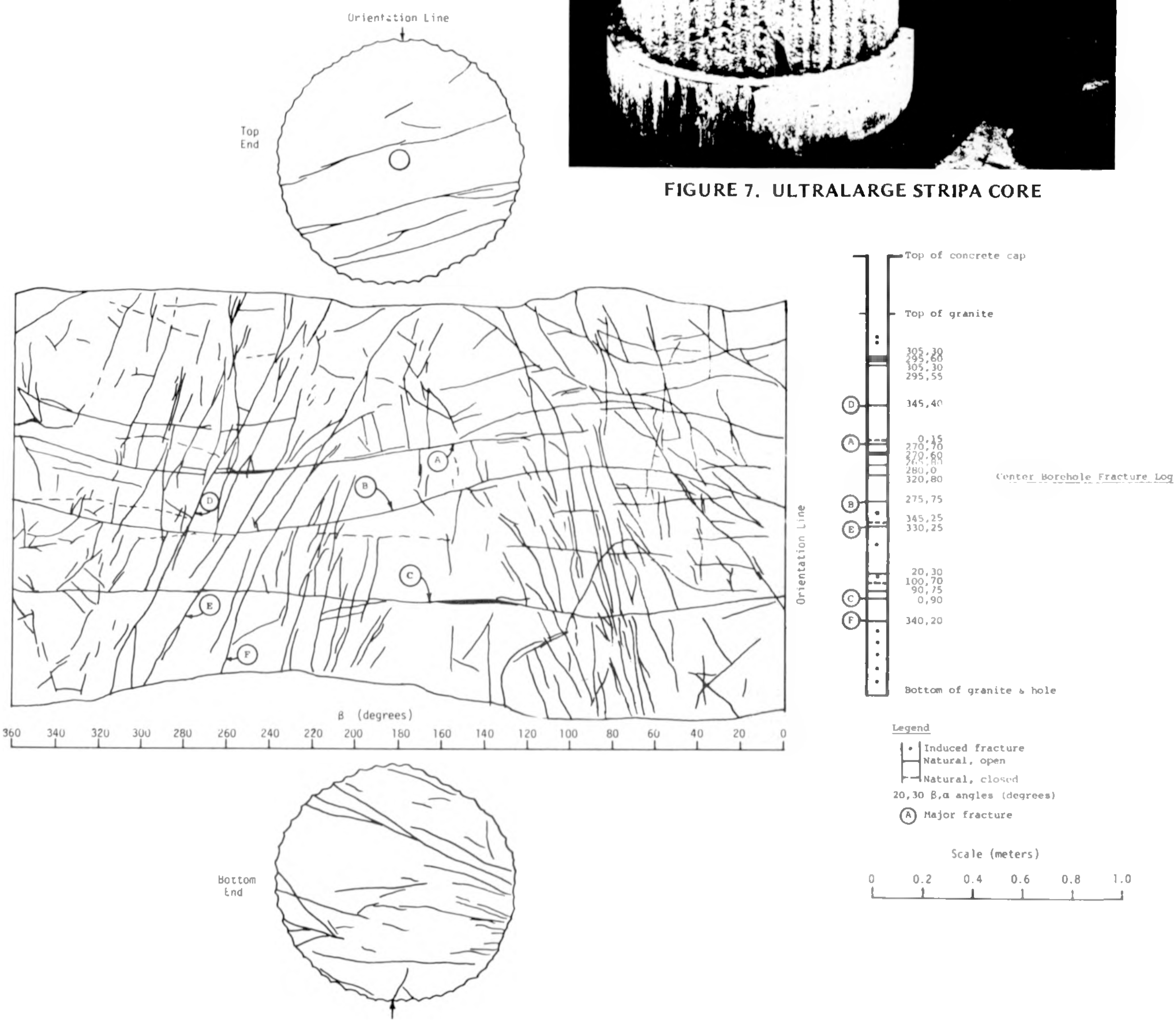

FIGURE 8. FRACTURES IN THE ULTRALARGE STRIPA CORE

OKLO NATURAL FISSION REACTOR

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The Oklo natural fission reactor program is a study of the transport of uranium decay products and fission products in geologic media for times as long as millions of years. These studies commenced in 1974, not long after French scientists announced the discovery of fossil nuclear fission reactors at the Oklo uranium mine in Gabon, Africa. These reactors operated at low power levels for about 5 x 10^5 years. Criticality was first achieved 2 x 10^9 years ago. Since that time, fission products formed during the reactor operation have been subjected to transport processes through the media surrounding the fossil reactors. The unique isotopic composition of certain U decay products and fission products permits the present investigation of stable end products of radioactive decay chains to infer information about the rates of transport of radioactive nuclides and the conditions that led to this transport. Such data should serve to validate models of reactor product transport in geologic media over long time scales. The development of such models, which are based largely on relatively short laboratory and/or field experiments, is an important part of the work to assess the safety of nuclear waste repositories. Results from the natural fission reactor program offer a unique way to validate the models over the long times that are important in assuring the public about the efficacy of geologic storage of nuclear wastes.

The main activities of the program during FY 1979 include: (1) determination of transport zones around the Oklo natural fission reactors using ^{206}Pb, (2) study of Ru and Tc migration around the Oklo reactors, (3) development of a U-Ru age-dating technique, (4) a comparative U-Pb and U-Ru chronologic study of the Cluff Lake U ore deposit, and (5) study of element migration paths around the Key Lake U ore deposits. Both deposits are in Canada.

^{206}Pb is produced by ^{238}U decay and ^{204}Pb has no long-lived precursor. Consequently, in rich U ore deposits ^{206}Pb/^{204}Pb ratio is large compared to the value in the surrounding rocks, and these ^{206}Pb/^{204}Pb data can be used to map transport paths of Pb escaped from the ore zones and, presumably, other nuclides as well. In addition, the Pb isotopic systematics can indicate the manner in which Pb escapes from its host U minerals. We have employed these principles to look at the distribution of Pb around the Oklo reactor zones. The conclusions are: (1) the escape of Pb from uraninite grains, including not only Pb produced by U decay, but also Pb incorporated into grains at the time of formation, is diffusion controlled, (2) samples enriched in Pb relative to U, contain Pb with an isotopic composition indicating that it is the Pb which diffused out of uraninite grains, and (3) the conglomerate layer which stratigraphically lies 3 to 5 m under the Oklo ore deposit is a transport path and repository of the escaped Pb.

^{99}Tc($T_{1/2}$ = 2.1 x 10^5 years) and Ru isotopes are produced by ^{235}U thermal neutron fission. The migration of these nuclides produced in zones peripheral to the Oklo reactors has been studied. Samples show both loss and gain of Ru and ^{99}Tc (now ^{99}Ru) relative to U. Samples enriched in Ru and Tc show greater enrichments of Tc than of Ru, suggesting the greater geochemical mobility of Tc.

All of the samples are from the same stratigraphic unit in which the reactor zones are located. The unit dips steeply and all the samples above the reactors are enriched in ^{99}Tc (now ^{99}Ru) relative to ^{101}Ru, whereas all the samples below show no Tc-Ru separation or are depleted in Tc relative to Ru. Also, the samples above are enriched in both Tc relative to U and Ru relative to U, whereas those below are depleted in both Tc relative to U and, with one exception, Ru relative to U. Since Tc decays to Ru, the relative separation of the two elements must have occurred within about 10^6 years (5 half-lives) after the formation of Tc. It, thus, appears that the hot, aqueous fluids moderating the reactors moved ^{99}Tc and Ru upwards, depleting the zones below the reactors in these elements relative to U and enriching the zones above the reactors, with the preferrential upward movement and retention of Tc. Ru and Tc appear to be moving distances of meters over periods of time as long as 10^6 years, although the maximum distances to which these elements were transported remains to be determined.

^{99}Ru, ^{101}Ru, ^{102}Ru, and ^{104}Ru are produced by spontaneous fission of ^{238}U, and, therefore, the Ru-U system can be used as a chronometer. The main difficulty, however, in developing this age-dating technique has been the quantitative assay of Ru isotopic abundances at the 10^{-9} g level using thermal ionization mass spectrometry. A marked increase in Ru sensitivity has been achieved, thus permitting the development of the method and the comparative U-Pb and U-Ru study at Cluff Lake, Canada. Cluff Lake was selected because of the good geologic control. The U-Pb data indicate an initial low-grade U mineralization at 1.33 $\pm$ 0.03 AE (AW = 10^9 years) with major mineralization events at 1.05 and 0.80 AE. The Ru isotopic data and U abundance data on two Cluff Lake ore samples yield ages of 1.32 $\pm$ 0.04 and 1.42 $\pm$ 0.05 AE, in reasonable agreement with the U-Pb age of 1.33 AE. The U-Pb data suggest that the actual mineralization of the two samples analyzed for Ru is 1.05 AE, and therefore, it appears that the U-Ru data record only the first mineralization episode.

Our plans for FY 1980 include analysis of additional Oklo samples in order to define more accurately element transport paths and rates. The Oklo deposit is in an argillite-sandstone sequence. For comparison we plan to continue our recently initiated study of element migration in crystalline rock; in this case in the high-grade metamorphic rock surrounding the Key Lake U deposit. Such a comparative study should provide valuable information about the migration of radionuclides on long time scales in two types of rocks being discussed as possible nuclear waste repositories.

GEOTHERMOMETRY AND DIAGENETIC STUDIES OF SHALES

C. E. Weaver
Georgia Institute of Technology

The first objective in this program was to perform a critical review of the data on the mineral and chemical alterations that occur during diagenesis and low-grade metamorphism of shale and other clay-rich rocks—conditions that under certain design assumptions, could be similar to those that might be expected from emplacement of heat-producing radioactive waste in a geologic repository. The second objective was to develop geothermometry methods to determine the thermal history of shales and to determine the geochemical, mineralogical, and structural changes which occurred in natural shales when they were subjected to burial temperatures in the range of 100 to 400 C.

Time as well as temperature will determine what phase changes will occur when shales are exposed to temperatures generated by high-level radioactive waste. For this reason it is necessary to study the diagenetic changes that have occurred in shales subjected to moderate burial temperatures for long periods of time. This consideration also requires being able to determine the maximum paleotemperature to which a shale has been exposed. Because of their high water content (20 to 40 percent) and relatively rapid reaction rates at low temperatures, smectites are the clay minerals most likely to be affected by heat. Water loss and mineralogic changes caused by an increase in temperature will increase the porosity and permeability of the shale. This should increase the rate of water movement and decrease the structural stability of the shale. After the pore water is removed, interlayer water will be released from smectites at temperatures that, depending upon design parameters, could be encountered in a repository for high-level radioactive waste. This water will be transferred to the pores and may initially cause an increase in porosity and plasticity. As the original and new pore water moves into or away from the repository, the thermal conductivity of the shale will decrease and the temperature increase; fractures may develop and settling may occur. Under such conditions, the structural stability of the shales in the immediate vicinity of the repository is likely to be weakened. Only minor chemical and mineralogical changes should occur in the dehydrated shale; however, where water is present, chemical and mineralogical changes (conversion to illite and chlorite) could begin at temperatures as low as 40 C and continue to temperatures in excess of 300 C. Some of these changes may take place over a period of a few years or less. These mineral transformations will increase the grain size of the clay minerals. This should cause an increase in porosity and permeability, facilitating the movement of water through the shale.

Permeability barriers and high-pressure zones may be created in shale when ions precipitate from solution. This may create problems, but in the long term could be beneficial. The heat generated by radioactivity will increase the solubility of the minerals and accelerate chemical changes. This should cause an increase in the ion concentration of the shale pore waters. The mobilized ions (Si, Mg, Fe, K, Ca, Na) will migrate along the temperature gradient and at various intervals precipitate and cause a decrease in the porosity and permeability of the shale and interbedded sandstones and siltstones. This could lead to the development of high pressures in the vicinity of the repository and increase the possibility of rock failure. As the temperature of the repository decreases the possibility of problems arising from the formation of high pressures should decrease. Ultimately, a permeability barrier should be beneficial because it will inhibit the flow of water into and out of the repository.

Illite-chlorite shales, largely of Paleozoic age, which have been subjected to burial temperatures in excess of 200 C should undergo only minor changes when exposed to heat that produces temperatures of 100 to 300 C. The heat will have relatively little effect on the adsorption properties of illite. Water problems will be minimal and structural strength should not be decreased.

A suite of Cambrian Conasauga illite-chlorite shale samples subjected to varying burial temperatures was collected from Georgia, Tennessee, and Alabama. Laboratory analyses indicate a large number of measured mineral and chemical parameters changed systematically with increased burial depth and, presumably, temperature. Some of these changes are: grain size increases; the number of expanded layers decrease; crystallinity of illite and chlorite increases; 2M/1Md ratio of illite increases; kaolinite and K-feldspar decrease; chlorite increases and becomes more segregated; illite replaces chlorite; illite flakes, and quartz grains become more angular; flake orientation increasingly deviates from bedding orientation; K content of illite increases; and Mg/Fe ratio of the chlorite increases.

K/Ar apparent ages vary systematically with burial temperature. The coarse size fractions have "too-old" detrital K/Ar ages and the fine fractions "too-young" ages, indicating the time (and temperature) of formation or uplift. With increasing burial temperature the K/Ar ages of all size fractions systematically approach the age of the finest fraction, indicating Ar has been progressively released from the coarser fractions with increasing temperature.

Oxygen isotope analyses of illite and quartz are expected to supply absolute temperature values to which the other parameters can be related. Preliminary data indicate the samples studied have been exposed to temperatures ranging from approximately 150 to 300 C. The apparent temperatures increase as the burial depth of the shale samples increase.

Future studies will be concerned with the analyses of additional samples to refine some of the trends that were determined during the initial analyses. Other measurements, such as cation-exchange capacity, porosity, surface area, density, cleavage, and rock strength will be made on a suite of samples exposed to various temperatures. The large volume of data will be analyzed statistically to develop a quantitative model that can be used to determine paleotemperatures and predict the changes that will occur when shale samples are exposed to an increase in temperature.

ROCK CHARACTERISTICS

THERMOMECHANICAL MODELING USING AN EXPLICIT NUMERICAL TECHNIQUE

R. Hofmann
Science Applications, Inc.

The rock mechanics activities of the San Leandro, California, office of Science Applications, Inc. (SAI/SL), for NWTS/OWNI during FY 1979 can be divided into five distinct tasks: (1) three-dimensional Project Salt Vault (PSV) simulations, (2) continuing support for the SAI rock-salt model, (3) simulations of the Stripa heater experiments, (4) development of a coupled thermomechanical-hydrology model for STEALTH 2D, and (5) participation in the nonsalt waste-isolation workshop.

These activities are primarily analytical and fall into two categories: (1) constitutive modeling and (2) multi-dimensional computer simulations. Within the scope of the first category, SAI/SL reviews laboratory and field data in order to develop an understanding for the complex constitutive mechanisms that may be important during the life of a geologic repository. In the context of the second category, SAI/SL provides expertise in the use of the explicit finite-difference (EFD) method, which is one of three fundamentally different techniques used within the ONWI community.

The EFD technique is well suited for handling complex nonlinear constitutive equations, large strain deformation, time-dependent response, and coupled physical mechanisms in one-, two-, and three-dimensional geometries. The specific EFD codes that are being used are called STEALTH*. They are in the public domain and were developed by the Electric Power Research Institute (EPRI) to perform consequence analysis for both design and regulatory requirements. In addition to PSV and Stripa, STEALTH is being used to simulate in situ tests, laboratory experiments, and design scenarios for the Waste Isolation Pilot Plant (WIPP) in New Mexico, to investigate design scenarios for the Gable Mountain experiment on the Hanford Reservation in Washington, and to perform earthquake safety studies of waste-isolation tunnels in generic geologies.

Technical Results

3D PSV Simulations. The three-dimensional simulation of PSV that was completed in FY 1979 represents a milestone in repository computations. The calculation was designed to (1) validate that STEALTH 3D is working, (2) gain experience in performing three-dimensional calculations, and (3) assess the economics of three-dimensional analyses. The results of the computations[1] were encouraging, but indicated that more calculations are necessary.

To validate that STEALTH 3D is a working code, a two-dimensional calculation of PSV was performed using STEALTH 3D, and the result were compared with previous STEALTH 2D results. Figures 1 and 2 display room-convergence data. The slight deviances between STEALTH 2D and 3D after standard day 1100 are due to an input error in which a creep variable was not properly reinitialized at restart time. Otherwise, the results of the two codes are quite close.

To gain experience performing three-dimensional calculations, a fully three-dimensional simulation of PSV was performed using the smallest possible computational domain. Figure 3 displays top, bottom, and side views of the computational domain. The results of this calculation, shown in Figures 4 and 5, are strongly influenced by the nearness of the haulageway boundary conditions. In retrospect, it is clear that this boundary was chosen too close.

Finally, an assessment was made to determine if three-dimensional calculations are economic. Although these first calculations have been moderately expensive, 3D analyses will only be a factor of 2 more costly (labor plus computer) than 2D analyses.

SAI Rock-Salt Model. The PSV calculations used a salt creep model proposed by Starfield and McClain[2] and modified by SAI/SL. The constants were calibrated to reproduce data from the heated-pillar experiments performed by Lomenick[3]. However since these experiments only provide thermally induced creep data, the material constants are probably not correct for regions of stress-induced creep. Recent creep data from Waversik and Hannum[4] and Hansen and Mellegard[5] on WIPP salt indicate that the material constants (in particular, the parameter "a")[2] are different for thermally induced and stress-induced creep. Using the value of "a" derived for thermally induced creep will result in excessive creep rates for creep due to low differential stress. Referring back to Figure 1, it can be seen that the constants developed from the Lomenick data do lead to excessive creep rates in the region in which thermally induced creep is essentially zero.

Stripa Heater Simulation. Currently, a series of STEALTH 2D and 3D simulations is being performed to model the heater experiments at the Stripa Mine in Sweden. Thermal-only and thermal-elastic calculations have been performed and the results have been compared to calculations performed by Chang, et al.[6] and Butkuvich[7] at LLL. The calculational results agree quite well with each other and the thermal-field measurements. However, none of the codes reproduced the experimental displacement history data because in these calculations, each code was constrained to use an elastic constitutive model. SAI/SL is currently repeating the thermomechanical calculations with a constitutive model called CAVS (Cracking and Void Strain) which includes the effects of joints and fractures. Figure 6 shows a conceptual flow chart of the CAVS constitutive model.

Hydrology Model for STEALTH 2D. A hydrology model based upon Darcy's law for flow in porous materials has been added to STEALTH 2D. Several one-dimensional test problems have been run in the uncoupled mode to test the updates. The results of these tests indicate that the model is working. A strategy for coupled calculations has also been worked out.

Future Research

Future activities are divided into salt-related and nonsalt-related efforts. For salt there will be three specific areas of activity:

- Continued evaluation of SAI creep model with respect to the available salt data; FY 1980 will concentrate on Avery Island data
- Recalculation of 3D PSV with the boundary in the haulageway moved into the pillar; also, modification of constitutive constants to reflect creep characteristics under low differential stress loads
- Evaluation of some generic design scenarios which involve 3D mechanisms such as (a) the asymmetric loading of a repository, (b) the influence of asymmetric bedding planes, (c) the stability of various shaft, room, haulageway intersection designs, and (d) horizontal emplacement.

In nonsalt-related activities there will also be three specific areas:

- Refinement of Stripa calculations by performing parametric studies
- Adaptation of CAVS to basalt for Gable Mountain design studies
- Investigation of CAVS/hydrology coupling scenarios.

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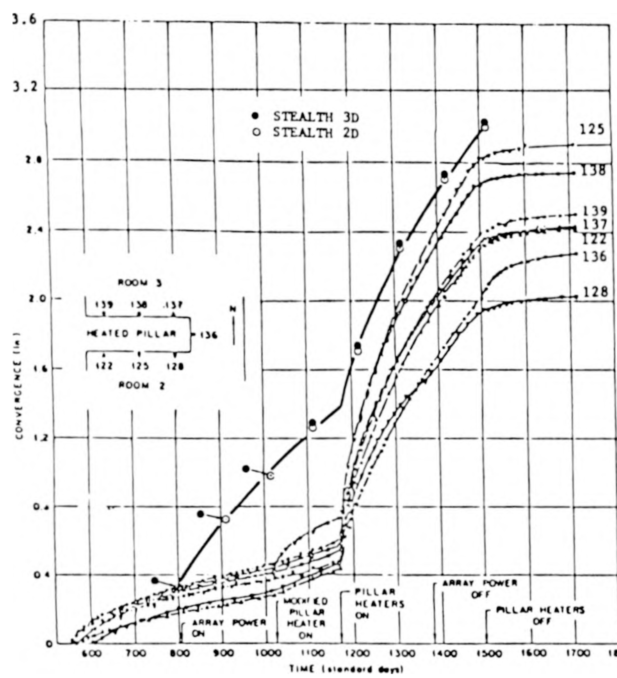

FIGURE 1. STEALTH 2D AND 3D, TWO-DIMENSIONAL VERTICAL CONVERGENCE DATA FOR THE PVS(3) HEATED PILLAR AS A FUNCTION OF TIME

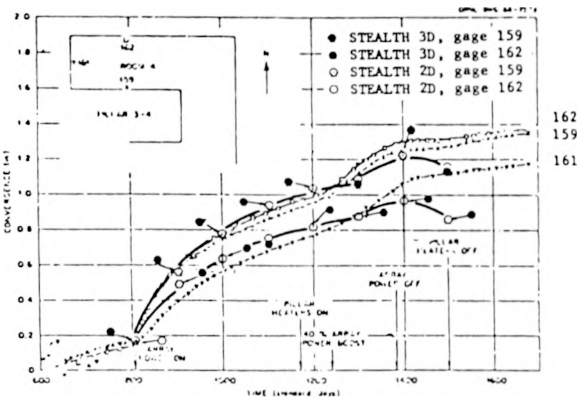

FIGURE 2. STEALTH 2D AND 3D, TWO-DIMENSIONAL VERTICAL CONVERGENCE DATA FOR PVS ROOM 4(3) AS A FUNCTION OF TIME (PIPE-TYPE GAGE)

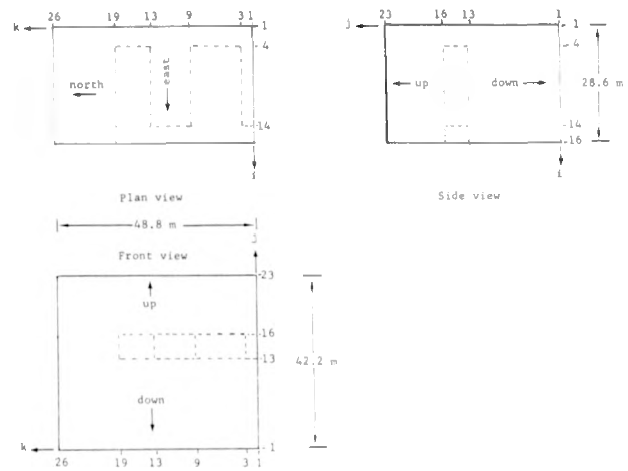

FIGURE 3. INITIAL STEALTH 3D MESH OUTLINE FOR THREE-DIMENSIONAL SIMULATION OF PSV

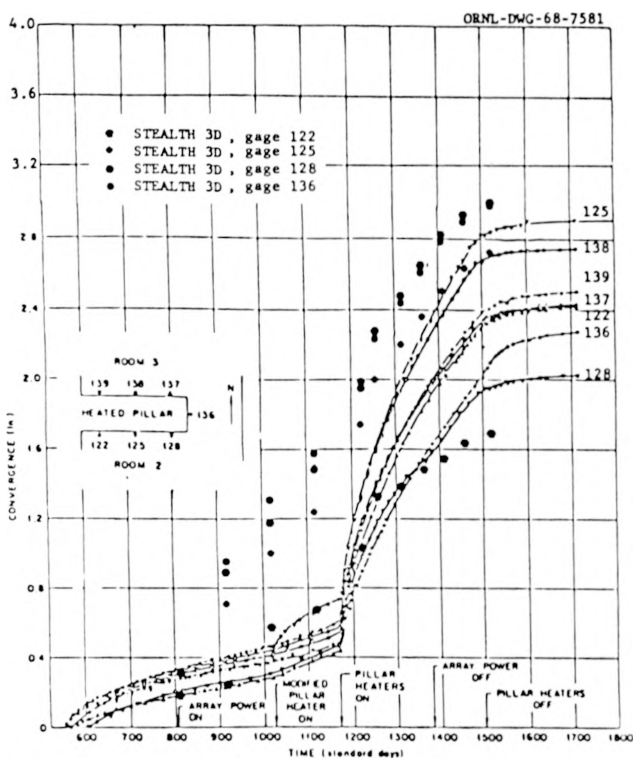

FIGURE 4. STEALTH 3D, THREE-DIMENSIONAL VERTICAL CONVERGENCE DATA FOR THE PSV(3) HEATED PILLAR AS A FUNCTION OF TIME

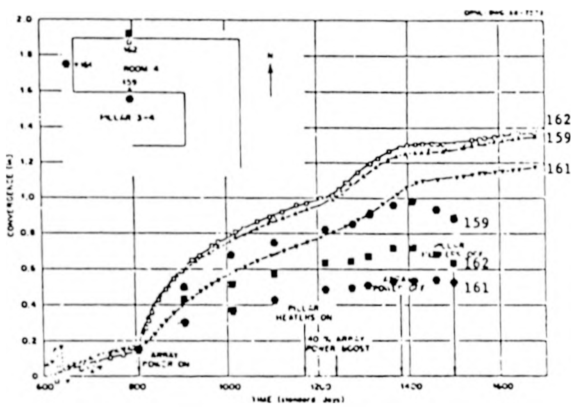

FIGURE 5. STEALTH 3D, THREE-DIMENSIONAL VERTICAL CONVERGENCE DATA FOR PSV ROOM 4(3) AS A FUNCTION OF TIME

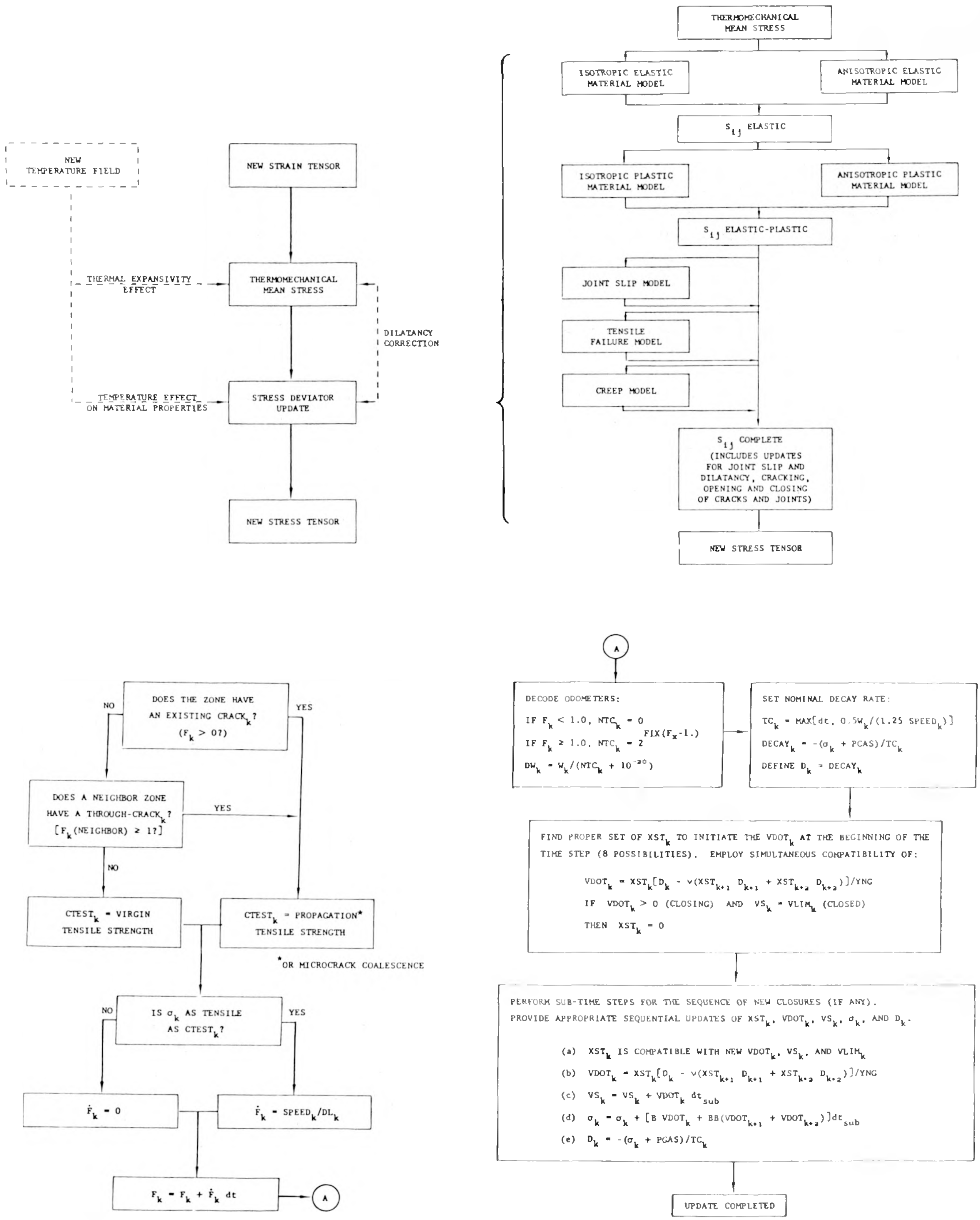

FIGURE 6. CONCEPTUAL FLOW CHART OF THE CAVS CONSTITUTIVE MODEL

THERMOMECHANICAL MODELING BY BOUNDARY-ELEMENT METHODS

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University of Minnesota

The general objective of this research program is to develop and apply various computer models for simulating the thermomechanical response of the rock around a nuclear waste repository. These computer models are based on a particular kind of "boundary-element method" called the "displacement-discontinuity method".[1] The displacement-discontinuity models are intended primarily as a means of predicting far-field effects of a nuclear waste repository. These models, therefore, complement other models based on finite-difference and finite-element methods, which at present are most useful for analyzing localized aspects of the problem.

Three basic computer programs for the displacement-discontinuity method have been developed: SALT, EXPAREA, and REPOSE. SALT is a two-dimensional representation of a repository, with the ability to model one-dimensional creep of pillars between rooms; EXPAREA is a similar three-dimensional model designed to analyze "small-scale" problems such as the experimental area of Project Salt Vault; and REPOSE is a large-scale, three-dimensional model designed to model a full-sized nuclear waste repository in its entirety. The capabilities of SALT and EXPAREA are described in References 2 and 3, respectively. The background for the program REPOSE is given in Reference 4.

The displacement-discontinuity models assume that the rock mass is linearly elastic, and that the nonlinear, inelastic behavior is confined to a single layer, or "seam", where deformations are related to time, stress, and temperature. Because of the assumed linear behavior of the rock mass, superposition is used to model 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 unmined portions of the seam; and (3) thermal-stress and displacement changes throughout the rock mass, including the seam, where stress and temperature changes affect the creep behavior of the seam material.

In the displacement-discontinuity method, the seam is treated as a number of elemental displacement discontinuities (or dislocations) whose values are determined such that the appropriate boundary conditions are satisfied at the plane of the repository. Boundary conditions, in turn, are governed by the overburden stress, the thermal-stress and temperature changes caused by the decay of the nuclear wastes, and the inelastic behavior of the salt in the seam. Nuclear waste is modeled as a series of heat sources with specified heat-generation characteristics. Creep deformation of the seam is computed from a creep law whose form was derived from laboratory-model pillar tests; and parameters for the creep law have been established from an analysis of the field data obtained from the Project Salt Vault experiment.[5,3]

During the past year (FY 1979), the above-mentioned computer programs were improved and made more comprehensive in preparation for repository design studies. The two-dimensional program SALT was generalized to allow for the possibility of slip or separation occurring along geologic discontinuities such as joints or faults.[1,6] The results of the SALT program also were compared with the predictions of the explicit finite-difference program STEALTH (developed by Science Applications, Incorporated) for a particular problem patterned after the Project Salt Vault experiment. This work confirmed that the bulk of the inelastic deformation indeed occurs in the pillars between rooms, as assumed for the displacement-discontinuity models, at least for the case that no heat sources are present. Further studies are planned to test the SALT program for more realistic problems involving heat sources.

Research for FY 1980 will consist of three separate tasks: (1) design studies using existing computer programs, (2) further computer-program development, and (3) laboratory experiments on the time, stress, and temperature response of salt. The laboratory investigations will be designed to complement studies being conducted by other contractors. The objective of the experimental program is to investigate certain aspects of the time-dependent behavior of salt that are crucial to the development and application of numerical models. Most of the work for FY 1980, however, will involve setting up the laboratory apparatus and verifying that it is functioning properly.

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THERMOMECHANICAL MODELING FOR REPOSITORIES IN GEOLOGICAL MEDIA

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RE/SPEC Inc.

The overall objectives of the technical effort by RE/SPEC are to assess the thermal/mechanical/hydrological behavior of repository situations in geologic media by numerical modeling (finite-element) procedures and to evaluate the deformational response of salt in the laboratory under applicable conditions of stress and temperature. The modeling effort is concerned primarily with the response of salt (bedded and dome) to the emplacement of heat-generating radioactive wastes, for very-near-field (global repository) geometrical situations and over time frames from excavation and emplacement through termination of the waste-induced thermal cycle. As input to the modeling activities, the development and refinement of constitutive laws for the deformation and failure of salt is the main goal of the laboratory effort. This modeling and laboratory work forms the thrust of a more general program at RE/SPEC in the area of waste-disposal technology, which includes in situ heater and brine-migration tests in dome salt at Avery Island, LA (ONWI), pre-WIPP and WIPP in situ experiment design and brine migration in bedded salt (Sandia), in situ experiment design in granite at NTS (LLL), and repository design and analysis for plutonic rock in Canada (AECL/Acres). The results of all of the above efforts in rock mechanics, as well as the interchange of technology developments, provide both basic and applied support to the more general NWTS/ONWI program in the areas of site evaluation, repository design, and safety assessment.

Significant Results from Current Studies

During FY 1979, modeling consideration has been given to the closure of waste-canister drillholes; disposal-room stability with and without crushed-salt backfill; influence of ventilation drifts on disposal-room stability; Avery Island in situ heater tests; and far-field thermal/mechanical/hydrological response of generic bedded and dome salt media to heat-generating waste emplacement. Drillhole closure and crushed-salt-backfill pressure on sleeves are of interest from the viewpoint of waste retrievability over a specified period of 25 years. Calculations indicate that the closure of 51-cm-diameter (20 in.) drillholes in salt over 25 years will not be detrimental to the retrieval of 30.5-cm-diameter (12 in.) high-level waste canisters for a gross thermal loading (GTL) of 37 W/m^2 (150 kW/acre) of high-level reprocessed waste. A similar conclusion was drawn for 41-cm-diameter (16 in.) spent-fuel canisters emplaced in 61-cm-diameter (24 in.) drillholes for a GTL of 32 W/m^2 (130 kW/acre). If the drillholes are sleeved and backfilled with crushed salt, the pressures generated on the sleeve should not be sufficiently great to warrant specially designed or fabricated sleeves.

Near-field modeling studies of disposal room stability indicate that:

- Room stability is most significantly influenced by repository depth over a GTL range of 7.4 to 11.1 W/m^2 (30 to 45 kW/acre) and a pillar-height-to-pillar-width ratio of 1:3 to 1:5.
- For minimum room closure, an optimum extraction ratio exists for a constant GTL at a specified repository depth.
- For the conceptual repository design in dome salt, the influence of the ventilation drifts on disposal-room stability is generally negligible.
- Disposal-room closure over 25 years can be significantly reduced by backfilling the room with crushed salt immediately after waste emplacement for a GTL of 8.9 W/m^2 (36 kW/acre), as indicated in Figure 1.

Far-field modeling studies of thermal/mechanical/hydrological perturbations around repositories indicate that:

- No intact-rock strength failure and only limited strength failure along preferred planes of weakness will occur in a generic bedded salt lithology for a GTL of 14.8 W/m^2 (60 kW/acre) of a 2:1 mixture of pressurized and boiling water reactor SURF, as shown in Figure 2.
- The time and magnitude of surface uplift for a repository in a highly thermally expansive salt bed is (1) relatively insensitive to depth for a given salt-bed thickness, and (2) quite sensitive to salt-bed thickness for a given depth.
- Perturbation of the regional groundwater flow system about a generic salt dome with a GTL of 37 W/m^2 (150 kW/acre) of high level waste subsides about 1,000 years after waste emplacement; the temperatures and thermally induced groundwater flow fields at 500 years are shown in Figures 3 and 4, respectively.

Future Studies

For FY 1980, consideration will be given to establishing the expected repository environment for a generic granite medium for high-level reprocessed waste, commercial SURF, and defense SURF; to stability evaluations for entry-entry intersections and inhomogeneous initial stress fields for repositories in salt; and to a preliminary assessment of a potential for thermal runaway under repository conditions. In addition, the complimentary laboratory effort for constitutive definition of the deformational behavior of salt will continue.

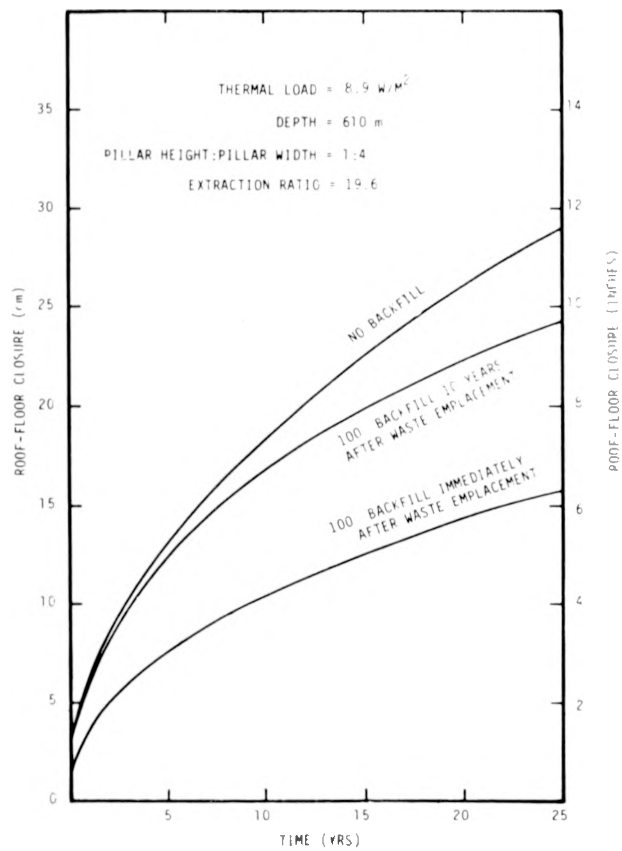

FIGURE 1. COMPARISON OF ROOT-TO-FLOOR CLOSURES AS A FUNCTION OF TIME FOR A ROOM-AND-PILLAR CONFIGURATION WITH AND WITHOUT CRUSHED SALT BACKFILL

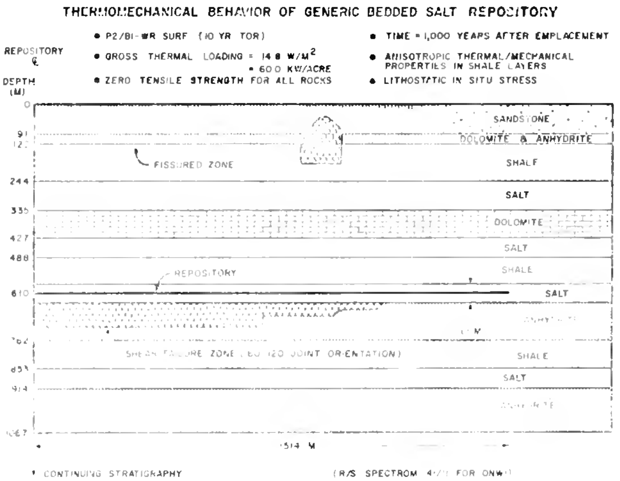

FIGURE 2. THERMOMECHANICAL BEHAVIOR OF A GENERIC BEDDED SALT REPOSITORY

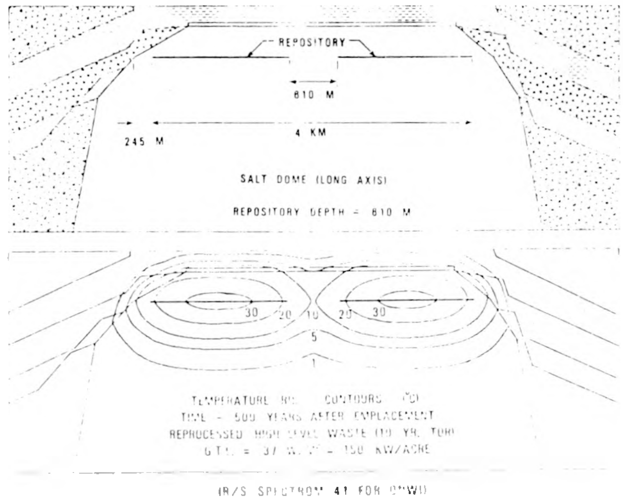

FIGURE 3. THERMAL BEHAVIOR OF A GENERIC DOME SALT REPOSITORY

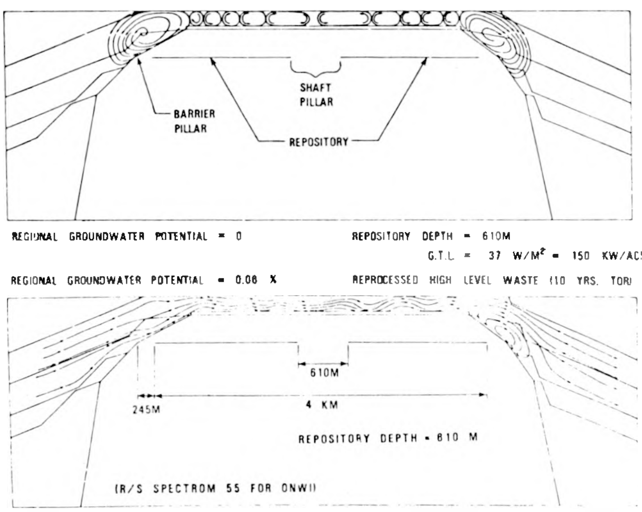

FIGURE 4. THERMALLY-INDUCED GROUNDWATER FLOW OVER A GENERIC SALT DOME AT 500 YEAR AFTER EMPLACEMENT OF HIGH-LEVEL REPROCESSED WASTE

ELASTIC, THERMAL, AND PERMEABILITY BEHAVIOR OF GENERIC REPOSITORY ROCKS AT IN SITU CONDITIONS

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Accurate prediction of both the short-term and long-term response of an underground repository for high-level radioactive wastes requires knowledge of the in situ stress/strain field, the permeability, the time-dependent elastic/inelastic response, and the thermal behavior of the rock mass. Additional information needed includes the geometry and depth of the facility, the heat load and decay characteristics of the waste, and the effects of temperature on permeability. A calculational model based on the chemistry and physics of the operative processes is also needed. In support of the goal, laboratory measurements are being carried out on several generic rocks over a range of lithostatic pressures, deviatoric stresses, and temperatures to be expected near a waste repository.

Elastic constants and thermal expansion of four coarse-grained igneous rocks are being measured at effective pressures to 50 MPa and temperatures to 300°C. These four are: Climax stock quartz monzonite (HTS), Stripa granite (central Sweden), Westerly granite (Rhode Island), and Creighton gabbro (Sudbury, Canada). Samples are 2.5 cm in diameter by 6.5 cm in length. Results of linear thermal expansion (α) measurements on the quartz monzonite indicate α ranges from 7 to 15 10^{-6} C^{-1} and is nonlinearly dependent on both pressure and temperature. Optical and SEM observations together with the intrinsic behavior of the component minerals suggest that crack porosity is responsible for the increased α at likely repository conditions.

Thermal conductivity, diffusivity, heat capacity, and α are being determined on Avery Island salt (Louisiana), Climax quartz monzonite, and certain other rocks at lithostatic pressures to 50 MPa, temperatures to 300°C, and pore fluid pressures to 25 MPa. Large cylindrical samples are being tested: 13 cm in diameter by 23 cm in length. Preliminary results on salt with no pore fluid pressure indicate a thermal conductivity of 6 $\pm$ W m^{-1} C^{-1} at 35 C, 10-40 MPa pressure. Diffusivity, heat-capacity, and α data have not yet been reduced.

Permeability, electrical conductivity, acoustic velocity, strain, and displacement are being measured on large fractured and unfractured samples of White Lake gneissic granite (Ontario, Canada), Westerly granite, and Sudbury gabbro. Lithostatic pressures range to 50 MPa, pore fluid (water) pressures to 25 MPa, and deviatoric stresses to 0.9 of the failure stress. The test temperature is 20°C, and the samples are 15 cm in diameter by 28 cm in length. Climax quartz monzonite, Montello granite (Wisconsin), and Stripa granite remain to be tested. Results for the initial three rocks indicate permeabilities of 10^{-16} to 10^{-20} cm^2 (10^{-8} to 10^{-12} Darcys), depending on rock type, pressure, and stress. Electrical conductance and acoustic velocity are well correlated with permeability and thus show promise as remote, nondisruptive indicators of permeability in situ.

TRANSIENT CREEP FLOW LAWS IN ROCK SALT

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The objectives of this project are to (1) develop a transient creep constitutive relation for rock salt based on operative deformation mechanisms, (2) measure experimentally the state variables required for evaluating transient creep theories, and (3) determine deformation mechanisms operative during transient and steady-state creep.

Constitutive relations for flow controlled by thermally activated processes provide the basis for extrapolating results of short-term laboratory tests at high temperature in order to predict deformation at the lower temperatures and over the longer times expected in radioactive waste-isolation repositories. Creep tests are being conducted on pure synthetic rock salt so that initial starting material can be controlled, and so that transient creep theories for pure aggregates can be applied to the test results.

Transient creep behavior is important in three time-temperature-space regions that are of concern in predicting the deformation associated with high-temperature radioactive-waste isolation:

(1) The temperatures (300 C) adjacent to a heat source may indicate that high-temperature transient creep may occur.

(2) Deformation adjacent to a heat source must approach a steady-state condition, so the relation between high-temperature transient creep and steady-state creep is critical to predicting long-term deformation at elevated temperatures.

(3) Creep mechanisms which lead to high-temperature transient creep in short-term laboratory tests may control steady-state creep at the lower temperatures and lower strain rates expected at greater distances from the heat source.

For these reasons, in order to predict the short-term as well as long-term deformation of rock salt surrounding radioactive materials, an adequate characterization of the high-temperature creep behavior of rock salt must be added to our present knowledge of the steady-state behavior.

Last year we examined transient-creep constitutive relations developed for metals and ceramics, and we formulated a transient-creep flow law which appears to be applicable to constant-strain-rate tests. We also have developed etching techniques which have enabled us to characterize a limited number of natural and artificial rock-salt aggregates.

Observational Effort

It is essential that the texture and fabric of the rock salt and the degree and type of deformation within the individual halite crystals be characterized in the synthetic rock-salt specimen before and after experimental deformation and in naturally deformed counterparts. This characterization permits detection of structural changes in the salt for controlled experimental conditions and the operative deformation mechanisms (slip and/or recrystallization). Comparisons between experimental specimens and natural counterparts provide a basis for extropolation of laboratory results to the field if it can be established that the same deformation mechanisms are operative in both materials.

Results to date are summarized as follows:

(1) Observational techniques for characterization of the texture and fabric of rock salt have been established and are employed routinely in study of experimental specimens.

(2) The *a*-axis fabric developed in synthetic aggregates is similar to that found in some natural rock salt. This rock fabric is characterized by one *a*-axis (001) lying perpendicular to bedding, while the other two *a*-axes are randomly oriented parallel to bedding.

(3) Review of the literature discloses that changes in the substructure of salt crystals do track changes in experimental conditions (e.g., Carter and Heard, 1970; Poirier, 1972), so that there is definite encouragement that our approach will be fruitful.[1,2]

Inclusions of two distinct types occur in most synthetic and naturally occurring rock-salt samples examined thus far. These presumably contain liquid or gas, but the exact nature or composition of the fluids has not been determined. The first type are intragranular, rectangular-parallelepiped-shaped fluid inclusions, that are crystallographically controlled with their sides parallel to (100), (010), and (001). These inclusions occur along planar zones or they are randomly scattered throughout the grain. They have a constant orientation within each crystal. Consequently, these inclusions can be used to determine the orientation of the crystallographic axes in each grain. In natural rock salt, the a-axis fabric determined from these inclusions is identical to that obtained from direct measurement of cleavage planes. Some of the inclusions have rounded corners and can look like bubbles, but, within any given grain, well-developed "cubic" ones can be found from which orientations can be obtained. Orientations determined in this and other ways are important because crystallographic fabrics of experimentally deformed synthetic rock-salt samples will be compared with naturally deformed rock salt. Similar fabrics imply that textures and deformation mechanisms in natural salt have been produced in the laboratory.

The second type of inclusions occurs primarily along grain boundaries and consists of irregularly shaped blobs and connecting tubes separated by regions of solid halite. With increasing annealing pressure or temperature, the connecting tunnels vanish, leaving the boundaries solely with isolated rounded bubbles.

Starting Materials

Three types of rock-salt specimens have been examined, namely: (1) synthetic rock salt produced from a granular aggregate upon application of confining pressure and temperature; (2) annealed synthetic rock salt; and (3) natural rock salt.

Annealed synthetic rock-salt samples obtained from Dr. H. C. Heard, Lawrence Livermore Laboratory, University of California, and used in his 1972 study[3], have an average grain size of 0.25 mm, and a size range from 0.18 to 0.73 mm. The grains are angular polyhedra, with most grains having four or more straight sides, but occasionally a grain with a curved side is observed. Triple-point grain boundaries are common, and the degree of deformation is low (low etch-pit density). Heard's material was reannealed prior to each of his experiments. Annealing at 400 C causes significant grain-size changes, indicating that the effect of grain size on the deformation of rock salt can be investigated experimentally by making starting material with different grain sizes through suitable annealing.

We are currently conducting a suite of systematic experiments to determine the appropriate temperatures and pressures for sintering and annealing synthetic rock samples in order to control the grain size.

Natural salt specimens from the Hockley salt dome, 18 km northwest of Houston, Texas, and from flat-lying beds in Lyons and Hutchinson, Kansas, have been examined in thin section. Except for local variations in grain size (ranging from 1 to greater than 40 mm) all of this material has similar petrographic and petrofabric characteristics. In comparison with Heard's annealed material, the natural salt is coarser grained, the grain shapes are somewhat more elongated, and the grain boundaries contain fewer bubbles. Subgrain size in polygonized grains is two to three times larger, and anhydrite occurs in distinct beds or in a dispersed manner among the halite crystals. The synthetic and natural material are similar in dislocation development, in the development of "cubic" fluid inclusions, and in the fact that the halite crystals are randomly oriented, crystallographically. Tentatively, we conclude that, except for grain size and shape, there are no significant differences between halite crystals and synthetic natural salt.

Transient Creep Theory

Theoretical expressions for transient creep under laboratory-imposed conditions of constant strain rate, constant stress, and constant stress rate have been derived. All are based on assumptions common to transient creep formulations of other workers: (1) deformation is caused by dislocation motion and diffusion-controlled recovery mechanisms; (2) first-order kinetics adequately describe the rate-controlling processes; (3) a constitutive equation can be derived from a scalar function of all state parameters, and the important state parameters are strain, strain rate, stress, stress rate, temperature, and structure.

The theoretical expression for transient creep in a constant-strain-rate experiment is :

$$\sigma(t) = \Sigma\, \sigma_i\, [1 - \exp(r_i t)] \qquad (1)$$

where σ_i are the steady-state stress values for particular deformation mechanisms and r_i is a rate parameter characterizing the rate at which the mechanism approaches a steady-state configuration.

To determine whether Expression (1) is valid for constant-strain-rate experiments, the parameters σ_i and r_i were obtained by fitting Expression (1) to Heard's (1972) data. Expression (1) predicts well both the steady-state stress values and the rate constant r_i for four experiments. Data for analyzing the effect that crystalline substructure may have on σ_i and r_i are not available. Our petrographic and petrofabric analyses of synthetic and naturally deformed rock salt are intended to provide some data for assessing the role that changing crystalline structure plays in transient creep.

Creep Tests

Our creep experiments will be conducted using artificially prepared aggregates of pure rock salt. The reasons for preparing our own rock salt rather than testing natural rock salt are twofold. First, the grain size of most natural rock salt is so large (up to 40 mm) that a 25 mm × 64 mm rock deformation specimen would contain too few grains. Second, although most natural rock salt contains impurities, most transient and steady-state creep theories assume monomineralic aggregates of pure halite. Therefore, proposed transient-creep constitutive laws must be evaluated on the basis of test data from pure rock salt. Once these theories are proven adequate for the relatively simple monomineralic aggregates, they can be modified for more complicated systems containing more than one mineral component.

Acknowledgments

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BENCH-SCALE CREEP TESTING OF DOME ROCK SALT

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Bench-scale creep testing of dome rock salt was conducted on rectangular blocks of salt. The blocks were testing in uniaxial compression. The tests were set up to provide deformation data on a bench scale. The geometry of the test, a circular hole perpendicular to a uniaxial stress field, was chosen to provide an easily modeled test case as a check against material models and computer codes which were derived from tests on smaller samples.

The blocks of salt for the test came from the Jefferson Island Company, at Jefferson Island, Louisiana. The samples were cut from the floor of the 1500-ft (457 m) level of the mine using an undercutter. Rough dimensions of the blocks were about 2.5 by 2.5 by 3.5 ft (760 x 760 x 1066 mm). These blocks were finished by the Rausch Granite Quarry of Ortonville, Minnesota. The blocks were cut to final dimension using a wire saw with brine for a cutting fluid. The faces were polished dry with special electric grinders. Final size for the blocks was 22 by 22 by 36 in. (558.8 x 558.8 x 914.4 mm). A 6-in.-diameter (152.4 mm) hole was drilled through the sample perpendicular to the long axis and centered and perpendicular to the rectangular face of the block. A thin-walled surface-set diamond drill was used, the cuttings being flushed with air.

The sample was loaded uniaxially, with the stress being applied by two mainfolds of three flat jacks each on each end of the sample. The stress was applied along the long axis of the sample. The load frame consisted of two steel platens, held together by eight 3-in. tie rods. The flat-jack manifolds were inflated by a high-pressure nitrogen bottle with a direct-relieving pressure regulator.

The test procedure was very simple. The sample was slowly loaded in increments to the ultimate load of 250 psi (1.72 Mpa). The instrumentation was closely monitored during the start-up.

The deformation was measured by four methods:

(1) Strain gages
(2) Dial gages
(3) Closure points
(4) DCDT displacement transducers.

In addition to this, the temperature and humidity of the test room were recorded.

The results of this testing were encouraging. Previous tests at higher stress levels were inconclusive because the salt rapidly failed in the tensile-stress zone at the surface of the hole. This crack soon spread the length of the sample, resulting in a test of not one but two independent samples. The newest data show no tension cracks after almost a year. The deformation rate, although fairly rapid at first, soon dropped to nearly linear rate, which seemed to vary little for the length of the test. Accelerated creep rate prior to failure was not seen at the low stress level. The physical appearance of the sample changed considerably during the test. Prior to loading, the sample was a uniform gray-white in color, with the grains being fairly indistinct. As the test progressed, areas of the sample, more or less in those areas of high stress, changed to a white color with the individual grains being more distinct. These grains seemed to have separated from the rest of the block, even though the sample did not fail.

LEACHING STUDIES

WASTE-FORM INTERACTIONS

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The purpose of a nuclear waste repository is to prevent radionuclides from reaching the biosphere. The multibarrier concept of containment has been referred to in numerous documents as a system which could be effective for this purpose. The waste form, however, ultimately functions as the "source" of the radionuclides (thus also supplying radiation and thermal fields) and as the initial engineered barrier to the release of radionuclides. It is the objective of Task 2 of the Waste Isolation Safety Assessment Program (WISAP) being managed by NWTS/ONWI to study the performance of candidate repository waste forms. To accomplish this objective, the work in this task has been divided into two phases:

Phase I Waste form-solution interactions
Phase II Waste form-system* interactions.

One must understand the behavior of the waste form-solution system, in the event of failure of the other barriers and to serve as a baseline for understanding the more complex reactions that will be studied in Phase II.

Studies have been going on for several years on waste form/generic solution interactions. This work will provide the functional relationships between element release, leaching time, solution composition and flow rate, and temperature. More detailed solid-state and solution analyses must be incorporated into these studies to gain knowledge of the mechanisms of release. Thus, in addition to element release rates, information on nuclide valence state and solution species must be provided. Solid-state analyses should include detailed mineralogic studies to identify alteration products such as recrystallized minerals. It is essential to understand release mechanisms if we hope to be able to predict long-term release from short-term laboratory tests.

Results to date from Phase I studies provide information on the effects of time, solution composition and flow rate, and temperature. Release rates decrease by a factor of 10 from both spent fuel and glass after 20 days of leaching. Another factor-of-ten difference is seen between a salt brine and a simulated bicarbonate groundwater, with the bicarbonate having the higher release rate. The release rates of actinides from glass increase an order of magnitude for flow rates going from 2cm/day to 60cm/day. Leach rates increase with temperature according to the Arrhenius relationship. Our data indicate an approximate doubling of the leach rate for every 10C increase in temperature.

Results on spent-fuel leaching to date indicate that congruent dissolution is the major mechanism of release, whereas for glass under similar conditions this is not the case. Another recent outcome of work on spent fuel indicates that, on low-flow-rate release scenarious, solubility limits, rather than kinetic phenomena, control the release of radionuclides.

The work on Phase II is in the planning and initial work stages at present and needs to be accelerated. The combination of waste-form leaching, mineralogic alteration of the waste form and surrounding environment, sorption, recrystallization, and precipitation reactions control the radionuclide release source term. This source term will be used for release-consequence analysis, and as input to far-field studies, thus making the connection from the waste form to far-field migration. Studies in this area will first combine the waste form plus canister plus host rock plus equilibrated groundwater using as many site-specific details and realistic repository conditions as possible. The effects of engineered barriers and backfill will be assessed as they are developed.

Phase II work will proceed from laboratory-scale (cm) to intermediate-scale (~1/2m) experiments. As much information as possible will be gained from tests that are easier to fabricate and are relatively inexpensive (laboratory scale) and to provide enough sample solution, waste form, and

*"System" = [waste form plus canisters plus engineered barriers plus host rock plus groundwater].

rock media for the solid-state, solution, and sorption/desorption studies and address the effects of scale-up (intermediate scale). Tests will be performed, as in Phase I, with both nonradioactive and fully radioactive waste-form systems.

Nonradioactive experiments using simulated wastes are used to get maximum solution, surface, and reaction-product characterization at minimum cost. Hot-cell experiments with actual waste solids enable the full spectrum of actinides and fission products to be studied. Also, the effects of a realistic radiation field can be investigated.

Common to all of the work outlined on waste-form interaction studies is the development of a model to describe the release of radionuclides from the waste package. The first model will be an empirical expression based on previous and ongoing work measuring radionuclides in solution as a function of the parameters listed earlier. As much mechanistic understanding as possible will be added to the model as more detailed solid-state and solution analyses are completed.

In addition to the laboratory- and intermediate-scale tests mentioned before, we feel that it is desirable to proceed with full-scale tests in a hot-cell facility. These tests would be designed to obtain the same information as discussed above and verify the results (and models) from the smaller scale tests. Tests would be made using full-scale canisters of waste surrounded with backfill/engineered barriers and host rock, simulating in-repository conditions. After a period of time the entire test would be dismantled and examined radially and axially. It is also feasible to incorporate accident scenarios into these full-scale tests.

In summary, WISAP Task 2 is an experimental program dedicated to providing nuclide flux data for release-consequence analyses and nuclide-migration studies. Task 2 will provide information for a release model and baseline understanding of release mechanisms. This information will be integrated with information coming from more complicated studies that have now started which couple together the near-field reaction zone. It is of extreme importance to gather and interpret this information so that repository safety analyses can be based on data that represent, as well as is possible, anticipated normal operating and accident conditions.

FISSION-PRODUCT RELEASE

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The objective of this project is to determine those conditions that serve to maintain the integrity of UO_2-type materials during underground storage for geologic times. The UO_2 ores at the Oklo natural fission reactor site in Gabon, Africa, have remained intact for about 2 x 10^9 years. Investigation of these ores [1] revealed that the lanthanide and actinide elements formed by the fossil reactors were retained on the whole within the UO_2 crystal structures. The determination of the conditions that resulted in the stability of the UO_2 can aid the NWTS program in its assessment of the suitability of proposed geologic sites for reactor waste storage. Furthermore, UO_2 in the form of spent fuel rods has been proposed as a nuclear waste disposal material. This work addresses aspects of the retention of fission products and actinide elements within UO_2 spent fuel rods under potential geologic storage conditions.

The major emphasis of this work is the effect of aqueous reducing conditions both on the rates of dissolution of finely divided UO_2 and ZrO_2 materials and on the rates at which reactor products are leached from spent fuel-element segments. In both of these sets of experiments, the effects of reducing conditions are compared with parallel measurements of the effects of oxidizing conditions. The results of the measurements of reactor-product leaching from the spent fuel elements will be correlated with the rates at which the UO_2 dissolves. Parameters whose effects on leaching rates will be investigated under controlled oxidizing or reducing conditions are temperature, hydrogen-ion concentration, common anion concentrations, and radiolysis effects.

This project commenced on March 24, 1978. Since that time, the work has resulted in a literature survey[2] of the diffusion, chemical, and morphological data concerning reactor products in UO_2 fuel elements; design and assembly of an apparatus for measuring the rate of dissolution of uraninite materials under controlled conditions; construction of a hot-cell slave box for use in the empirical measurements of fission-product release from nuclear power reactor spent fuel; and procurement of the spent fuel.

The first experiments to measure the dissolution rate of a material with very low solubility were performed with ZrO_2. The zirconium in ZrO_2 does not exhibit multiple oxidation states. Thus, its rate of dissolution should be independent of whether the solution is in a reducing or an oxidizing condition. A readily available tracer ^{95}Zr, with a convenient half-life of 64 days, enables the use of sensitive radiochemical techniques to determine small zirconium concentrations. Also, ZrO_2 is one of the materials being proposed as a part of a synthetic rock for use as a radioactive waste isolation host. When ZrO_2 was prepared in the fluorite crystal structure by the inclusion of 12.2 mol percent CaO (the same crystal structure as that in which UO_2 exists) the rate of ZrO_2 dissolution in deionized water at 25 C was measured to lie within the limits of 10^{-8} to 10^{-10} g cm^{-2} day^{-1}. A value for the rate, rather than a range, will be determined on receipt and installation of equipment that will permit measurement of the surface area of the radioactive ZrO_2 samples. Experiments are under way to measure the dissolution rate of pure ZrO_2, which exists in the monoclinic crystal form, but the results are not available yet.

Three experiments have measured the rate of dissolution of UO_2 in deionized water. In one experiment under reducing conditions, the Eh was maintained at -0.15 V at a pH of 4.2 and a 70 C. The measured uranium concentrations in solution remained constant at about 50 parts per billion (ppb) for the first 25 days, then rose to about 109 ppb for the remaining 29 days of the experiment. Additional work is being undertaken to help us understand these results. For the two experiments under oxidizing conditions, the Eh was +0.48 V at a pH of 4.1. The temperature for one experiment was 70 C. For the other experiment, the temperature was 70 C for 7 days and 25 C for 5 days, but the data indicate that the dissolution rate appears to be independent of temperature over this range. The UO_2 dissolution rate was determined to be 9×10^{-7} g cm^{-2} day^{-1} for the 70 C experiment, measured over 54 days, and 6×10^{-7} g cm^{-2} day^{-1} for the other experiment.

Experiments to measure the rates of reactor-product release from UO_2 spent fuel elements have been performed with two pairs of Zircaloy-clad fuel segments from H. B. Robinson-2 reactor. One wafer in each pair was leached under oxidizing conditions (Eh = +0.44 V at a pH about 5), while the other wafer was leached under reducing conditions (Eh = -0.12 V at a pH about 5). A pair of wafers was leached in deionized water for 65 days at 20 C. By that time, the leach rates appeared to have stabilized. The temperature was raised to 70 C. The available data show that about ten times more ^{90}Sr, lanthanides, and actinides were leached under oxidizing conditions than under reducing conditions. The quantities of ^{137}Cs leached were about the same under the two sets of conditions. When the temperature was raised to 70 C, the rates of leaching of the lanthanide and actinide elements decreased, compared to the rates at 25 C. In contrast, the data for ^{90}Sr, ^{125}Sb, and ^{137}Cs showed increased rates of leaching initially at 70 C. This observation may indicate that these species are more likely to be concentrated at grain boundaries in the UO_2, and thus be more easily leached than the lanthanide and actinide elements which are contained mainly within the grains or the UO_2 crystal structure. The data available from this spent-fuel leaching experiment are shown in Figures 1 through 4. The cumulative data are plotted at the end of each leaching interval, while the differential data are plotted at the midtimes of the leaching intervals.

The second pair of spent-fuel wafers was leached at 25 C to serve as a duplicate of the leaching experiments with the first pair of wafers. The data are remarkably similar to the first set.

In all of this work, the differences between oxidizing and reducing conditions amounts at most to an order of magnitude both in the rates of dissolution of UO_2 and in the rates of leaching nuclides from spent fuel elements. These differences are much less than we expected. We plan to determine whether larger differences can be observed under different experimental conditions. Finally, we want to learn why our expectations differ so much from our preliminary data.

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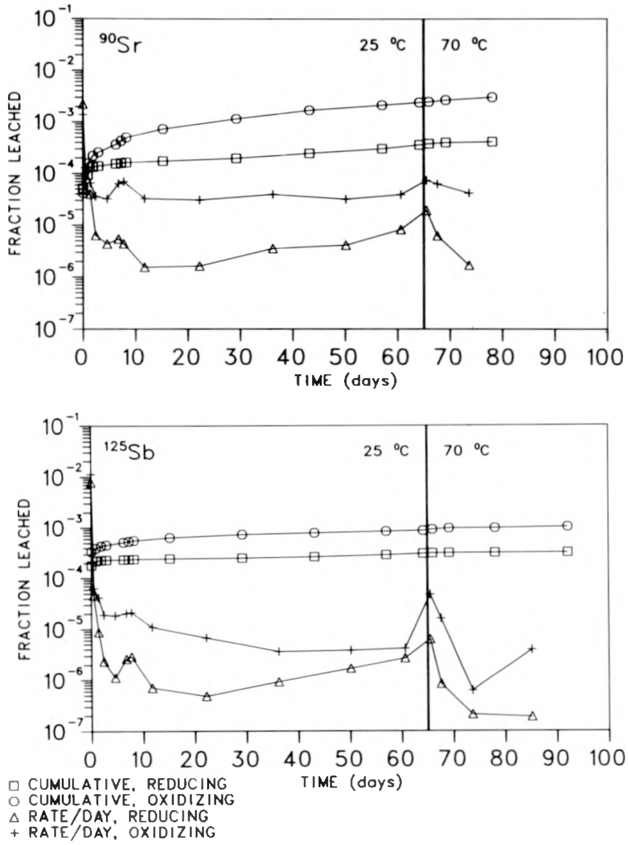

FIGURE 1. SPENT-FUEL LEACHING-EXPERIMENT ^{90}Sr AND ^{125}Sb DATA

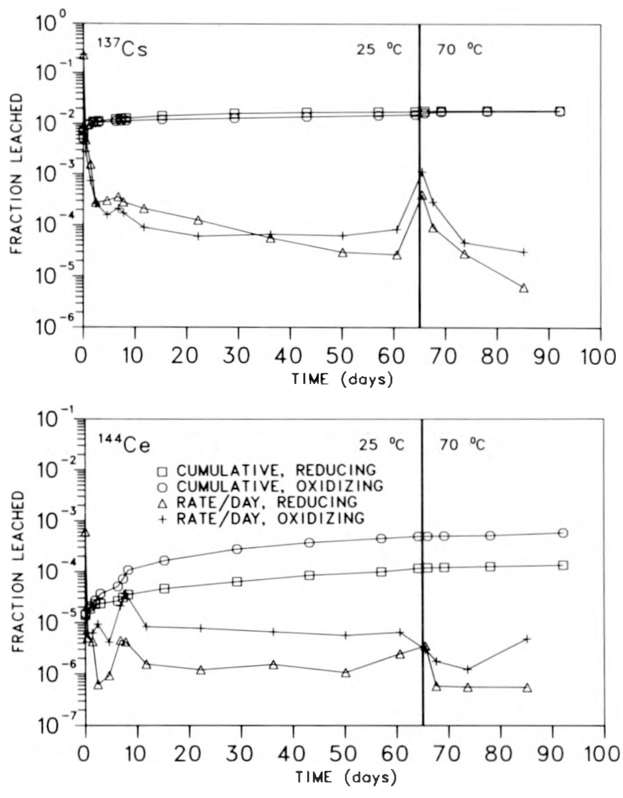

FIGURE 2. SPENT-FUEL LEACHING-EXPERIMENT ^{137}Cs AND ^{144}Ce DATA

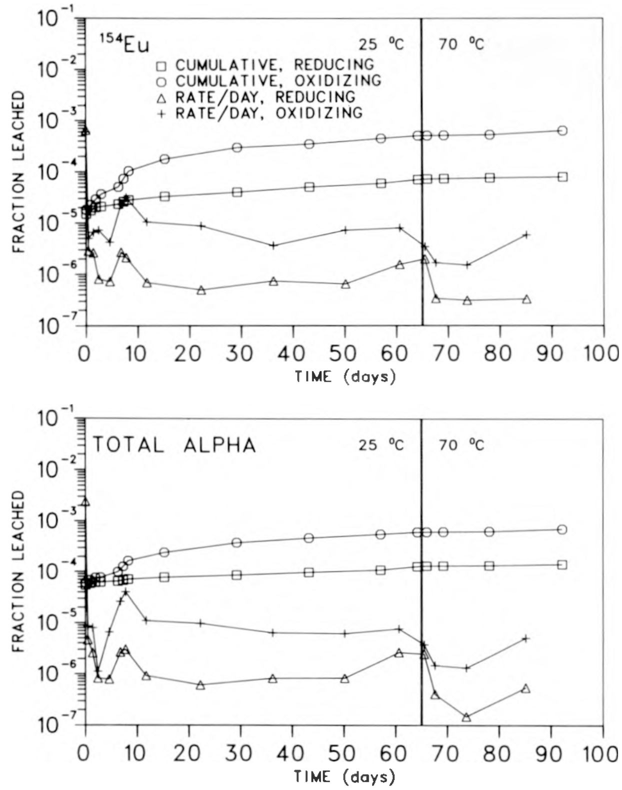

FIGURE 3. SPENT-FUEL LEACHING-EXPERIMENT ^{154}Eu AND TOTAL ALPHA DATA

Total alpha activity is a measure of ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, and ^{244}Cm.

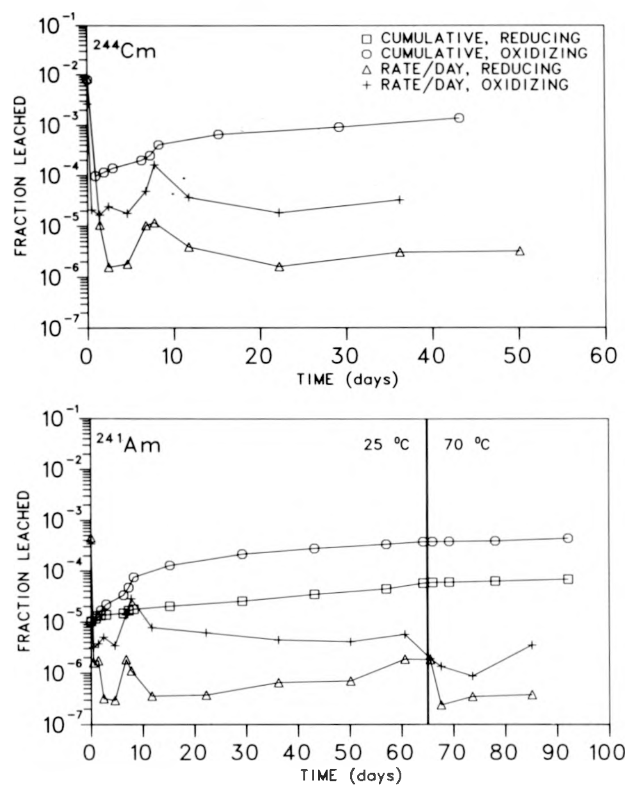

FIGURE 4. SPENT-FUEL LEACHING-EXPERIMENT ^{244}Cm AND ^{241}Am DATA

WASTE-ROCK INTERACTIONS

THE PHYSICOCHEMICAL PROPERTIES OF BITTERNS IN ROCK SALT APPLIED TO THE DESIGN OF RADIOACTIVE WASTE REPOSITORIES

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An understanding of the nature of chemical and physical interactions between host rock salt, canister, wastes, and any fluid present must be obtained to design for maximum waste containment within a mined repository in salt. These data are needed both to determine the initial conditions of waste emplacement in the repository and to estimate the higher temperatures and pressures that will be attained after the repository has been loaded and sealed. A thermal pulse peaking many tens of years after waste emplacement and lasting hundreds of years is being anticipated in the design of repositories. The maximum temperature to be attained over tens of meters of salt host will probably be kept below 100 C by the choice of canister spacing or thermal output. However, surface temperatures of the canisters may be as high as 300 C. Brines in the rock salt tend to be mobilized by thermal energy, either by fracture-decrepitation mechanisms, by migration of fluid inclusions, or by dehydration of hydrous minerals. The presence of brines can be expected to increase the possibility of canister corrosion and waste leaching, to weaken the rock strength, and to decrease sorptive properties of the host rock. One or more of these factors could affect the retrievability of wastes, and at the worst might threaten the integrity of waste containment. Hence, the focus has been to characterize the physicochemical properties of the fluids and solids in the system Na, K, Ca, Mg//C1, SO_4, Br and their reaction products from interaction with components of the integrated waste disposal system. These data will help in the design of the waste disposal system, water getters, and radionuclide adsorbers.

Several workers[1-3] have developed and applied equations for calculating the migration rate of fluid inclusions under a thermal gradient in rock salt. The results vary widely from as little as 7 liters to 43 liters in 10 years. All the models are based on a diffusional mass-transport mechanism rather than a mass-transport mechanism based on convection, which would yield significantly higher migration rates. Part of the cause for the scatter in the results based on the diffusional mass-transport mechanism is that many of the data required for the calculations are not available. Hence, the procedure followed by the various workers has been to estimate the requisite parameters. Two of these parameters are the density of the saturated solution (ρ_s) and $\partial_x/\partial T$, where χ is the concentration of NaCl in the brine. Measurements of ρ_s are completed for NaCl-H_2O up to 104 C and of $\partial_x/\partial T$ for a group of brines up to 100 C and up to 270 C for WIPP-A.

Prior to these measurements, those using the NaCl-H_2O system to approximate the fluids in the inclusions[1,2] obtained their ρ_s values from the 1904 data of Berkeley[4] except for Gaffney[3] who used a correlation by Haas[5]. To obtain higher temperature data, Bradshaw and Sanchez[1] extrapolated the older data set[4] linearly up to 300 C. The two literature sets for ρ_s do not agree where they overlap. Our experimental measurements of ρ_s for NaCl-H_2O continued the values of Haas[4] within the tolerances of the correlation. The differences in the ρ_s values would yield migration rates a few tenths of a percent higher than the ρ_s data used by Bradshaw and Sanchez[1]

The $\partial_x/\partial T$ values of some representative solutions measured in this study are listed in Table 1 as a function of temperature along with the $\partial_x/\partial T$ values for NaCl. The values of $\partial_x/\partial T$ are significantly greater than those of NaCl. Hence, if these values are used in the migration equations the results will be significantly greater migration rates than if the $\partial_x/\partial T$ values for NaCl are used.

The field experiments designed to measure the amount of brine that will migrate to a heat source in salt as a function of time are measuring that brine movement by collecting steam and weighing the amount of condensate. The weight of condensate is then assumed to be equal to the amount of the brine that has arrived at the heater. However, the amount of condensate that is collected does not represent the total amount of brine, but rather only part of the brine, because the brine is composed of nonvolatile salts and water, whereas the steam condensate is nearly pure water. Only a fraction of the water in the brines can be converted to steam. As the brine boils it becomes saturated with solid phases that contain structural water, and the boiling point of the brine is elevated. The total weight of bittern arriving at the heater is related to the weight of steam condensate as follows:

$$\Sigma_b = c \cdot N \quad (1)$$

$$c = \frac{1}{x \cdot f} \quad (2)$$

where Σ_b is the total weight of bittern arriving at the heater, N is the weight of condensate collected, c is a constant for each variety of bittern, f is the weight fraction of water in the bittern, and x is the weight fraction of the water that can be converted to steam.

The value of c is unity only for pure water. In the case of a simple salt where *x* is approximately unity due to the lack of hydrates and a low boiling-point elevation, the only correction required is for the amount of salt dissolved in the brine. For example, for sodium chloride at 200 C with a solubility of 31.898 weight percent, c is equal to 1.468. However, for bitterns which can have very high boiling-point elevations and form highly hydrated salts, the value of c is significantly larger. We have determined values of c directly as a function of temperature and bittern composition. This was accomplished by heating a known mass of bittern of specific composition in a sealed autoclave at a known temperature and weighing aliquots of steam that were removed through a valve assembly. This procedure was continued until no more steam could be extracted from the autoclave. In general, the value of c was found to range from about 1.7 to 10, depending on the temperature and the composition of the bittern.

These types of experimental data can be applied to the design of repositories in salt by allowing for more accurate calculations of the migration rates of fluid inclusions. The resultant theoretical predictions can be applied to short-term heater tests whose data collection can be improved using the c values. The results can be used to verify or modify the theoretical approach to permit long-term predictions with respect to brine inflow around waste canisters.

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TABLE 1. $\partial_x/\partial T$ in g/100 g OF SOLUTION °C^{-1}

		$\partial_x/\partial T \times 10^3$		
T degrees, C	**NaCl-H_2O**	**WIPP-A**	**78 percent $MgCl_2$**	**15 percent $CaCl_2$**
25	12.5	38.7	22.7	23.0
50	17.8	34.1	28.0	28.3
75	23.1	33.5	33.3	33.6
100	28.4	36.8	38.6	38.9
150	39.0	55.3	—	—
200	49.6	89.6	—	—
250	60.2	139.7	—	—

BRINE-WASTE FORM INTERACTIONS UNDER MILD HYDROTHERMAL CONDITIONS

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The multibarrier concept of nuclear waste storage requires information on the stability of waste forms in the geologic environment, the reliability of the containment (canister plus overpack), and the effectiveness of the geologic formation in blocking migration of dangerous radionuclides. The geological environment consists of the minerals that make up the repository wall rock and the fluids either contained in the rocks or circulating through them. The objective of this study is to determine the chemical reactions that might take place and the product materials formed, in the event that the containment is breached and waste forms come into contact with fluids and wall rock.

The bulk of the work in the program has considered three prototype waste forms: a borosilicate glass (PNL 76-68), a tailored ceramic (Supercalcine formulation SPC-4), and several simulations of spent fuel (UO_2 alone, UO_2 with 3.5 percent fission-product elements, and UO_2 with 40 percent fission-product elements). The formations are bedded salt and a suite of argillaceous rocks (shales and mudstones) exhibiting a range of chemical behavior. Fluid compositions range from nominal silicate rock groundwater to high-magnesium brine. The role of water is crucial: as a component in chemical reactions, as a catalyst for reaction kinetics, and as a transport medium. The greatest reactivity is expected during the thermal period (first few hundred years after burial when the canisters produce significant quantities of heat). Since actual repository temperatures will be determined by repository design, waste loadings, and time in storage before burial, the temperature range of 100 to 400 C has been investigated. Waste forms, fluid, and wall-rock materials were reacted in closed capsules for times of one week to one year under a nominal pressure of 30 MPa (300 bars).

Some of the important results are summarized in Tables 1 and 2. Details appear in a series of reports [1-4] and some have appeared in the open literature [5, 6]. It can be seen that the chemistry of the fluid is of great importance and that the reaction behavior is quite different in groundwater and in the high-magnesium "bittern" brine.

In Table 1 it is seen that the bittern brine attacks spent fuel and glass at 200 C and Supercalcine ceramic at 300 C with the effect that major amounts of numerous hazardous elements (or their analogs, e.g., Ln for Am, Cm) are released into solution. The results indicate that for any of these waste forms, hydrothermal conditions should be avoided by lower thermal loadings of the waste and/or the repository or that these forms should be protected from contact with hydrothermal brines throughout the thermal period.

Table 2 summarizes the stability trends for the four waste forms in solutions typical of silicate rocks. In these closed-system experiments, the major phase in spent fuel, uraninite (UO_2), remained largely unaltered, but substantial amounts of the fission products Cs, Rb, and Na could be extracted by contacting solutions. Trace levels of U were also taken into solution. When the solutions were also in contact with basalt or shale, the extracted elements were "fixed" in the solid by reaction to form the mineral-like phases pollucite and powellite.

The alteration of the reference glass in deionized water and a typical silicate groundwater at 300 C has already been reported in detail.[5] Yet, in presence of low-Eh solutions in contact with silicate rocks, waste-rock interactions could convert the glass-rock composite into a suite of new mineral-like phases. Results to date suggest that Cs and Sr are removed from the solution resulting from the interaction of water and glass by precipitation as phases related to the minerals pollucite and feldspar. For these two elements the interaction products are less soluble than the original waste form.

Shales and other argillaceous rocks vary widely in effective redox potential due to the variable presence of iron oxides, sulfides, and carbonaceous matter. The uptake of uranium, in particular, is much lower in systems containing these reducing agents. The low oxygen activity serves to hold uranium as the insoluble UO_2. The same effect is expected with respect to transuranics and other variable-valence elements. Such geochemical factors are of particular importance in the storage of spent fuel.

It is interesting to note that, with either the glass or the ceramic, the end result of hydrothermal conditions in the repository could be quite similar when viewed on the scale of the immediate repository. The glass could interact strongly with the silicate rocks to form mineral-like substances. The Supercalcine ceramic could interact only slightly. In both cases, the resulting solutions could contain little of the radionuclides of concern.

Unanswered questions requiring further research include: (1) chemical reactions involving intermediate brine compositions; (2) the role of accessory minerals, particularly oxygen buffers, in the wall rock in restricting radionuclide migration by maintaining reduced valence states; (3) the distance scale between a zone of waste-dominated chemistry in the immediate contact zone of the canister and a rock-dominated chemistry that must occur short distances into the repository wall.

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TABLE 1. HYDROTHERMAL REACTIVITY TRENDS IN BITTERN BRINE

Waste Form	Alteration of Solid	Elements in Solution
Spent fuel (200 C)	Some reaction. Under investigation	Major Cs, Sr, Ln, Ba, Rb, U Minor Mo
Reference glass (PNL-76-68) (200 C)	Not crystallized Solid intact	Major Cs, Sr, Ln, B, Rb, Ba, Zn Minor U, Mo, Si, Ni Trace Fe
Prototype crystalline ceramic (SPC-4) (300 C)	Breakdown of apatite, pollucite, and scheelite Monazite, fluorite and hematite stable	Major Cs, Sr, Ln, Rb, Ba, Ni Minor Cr, Si Trace Fe
Coated ceramic (300 C)	Some recrystallization of Al_2O_3 outer layer	No detectable Cs or Rb

Ln = Lanthanides (La, Ce, Pr, Nd, Sm, Gd, Y)

TABLE 2. HYDROTHERMAL REACTIVITY TRENDS IN DEIONIZED WATER AND "SILICATE GROUNDWATERS"

Waste Forms	Alteration of Solid	Elements in Solution	Waste-Rock Interaction Effect
Spent fuel (200 C)	Uraninite (UO_2) Stable	Major Cs, Rb, Mo Trace U	Cs, Rb, Mo "fixed" by interactions UO_2 stable in low Eh solutions
Reference glass (PNL-76-68) (300 C)	Complete	Major Na, B, Mo, Cr Minor Cs, Si, Ni, Rb, Ca Trace U, Ln, Sr, Ba	Most elements in solution "fixed" by interactions New mineral-like phases
Current crystalline ceramic (SPC-4) (300 C)	None[a]	Minor Na, Rb, Mo, Si Trace Cs, Sr, Ba, Ca	Released elements "fixed" by interactions Solid-solid interactions
Coated ceramic (400 C)	None	No detectable Cs or Rb	None

[a]Crystallinity of synthetic minerals enhanced; noncrystalline solids have crystallized.

STATUS REPORT ON SORPTION-DESORPTION PHENOMENA

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The work on nuclide sorption-desorption phenomena is being performed to (1) develop the capabilities needed to assess the postclosure safety of geologic repositories, (2) obtain scientifically defensible generic and site-specific data necessary for safety assessments, and (3) assist NWTS in site selection and licensing requests.

The PNL Sorption/Desorption Analysis Task (conducted through FY 1979 as WISAP Task 4) is investigating radionuclide sorption processes. If radionuclides are actually released into a transporting groundwater, these may be sorbed by the geomedia which they contact. Irreversible sorption would act to remove the radionuclide from the water. Reversible sorption would act in a manner similar to waste-form leaching, by providing time delays to the migration of radionuclides. Geomedia of sufficient sorptive capability could provide isolation of the waste from the biosphere by extending transit times to very long periods of time. As with leaching, sorption/desorption by geomedia is dependent upon the specific radionuclide involved and the physicochemical characteristics of the geomedia and transporting solutions. The objectives of the Sorption/Desorption Analysis Task are to:

(1) Investigate the fundamental phenomena governing sorption/desorption of radionuclides by geomedia

(2) Provide measured values of sorption distributions for specific nuclides and media

(3) Develop predictive equations for sorption distribution extrapolations for nonmeasured situations and extended time frames.

To address and work toward meeting the objectives, the specific activities of the task include:

- Evaluation of sorption/desorption measurement methods and development of standardized measurement procedures
- Laboratory measurement of the sorption/desorption behavior of significant long-lived

radionuclides in contact with a wide range of geologic media and groundwaters

- Statistical analysis and synthesis of these data to provide information on which variables correlate best with the observed sorption/desorption phenomena and to allow empirical interpolations to be performed within the environments studied
- Performance of verification studies to assess the degree to which short-term laboratory studies predict long-term in situ behavior
- Development of a more fundamental understanding of sorption/desorption processes to permit prediction of the fate of nuclides at a specific site from data available on other similar sites or from commonly measured parameters
- Tabulation and interpretation of input data for support of licensing efforts.

Currently this task includes work at PNL and at 10 subcontractors to generate the sorption/desorption data necessary for safety-assessment calculations. Details of past years' work may be found in WISAP Annual Technical Progress Reports.[1, 2]

For the past year, efforts have focused on the following with significant accomplishments noted.

Various experimental methods are being evaluated to measure the distribution coefficient (Kd) of radionuclides onto geologic media. This work is currently approximately 60 percent completed, with the goal of identifying standard methods by the end of FY 1981.

Although no one method of Kd determination has been identified as being superior, some of the past variability noted when comparing results from various methods has been identified as being related to solids-liquid separation techniques, tracer concentration and groundwater spiking techniques, container sorption correction, and subtle changes in pH, Eh, and groundwater compositions. Work to date has also led to standardization of more detailed characterization of rocks, groundwaters, and experimental conditions, such that variability in Kd results may be better explained. Although a standard sorption method is not at present available, a standard protocol for media characterization and Kd reporting has been identified.

Kd's were generated for a range of rocks and minerals contacting groundwaters and salt brine. Minerals utilized include: quartz, albite, anorthite, microcline, hornblende, enstatite, augite, illite, kaolinite, montmorillonite, vermiculite, biotite, attapulgite, apatite, and magnetite. Rocks that were studied include: shales, granites, basalts, limestones, dolomites, anhydrite, glauconite, tuffs, sandstones, and unconsolidated sedimentary sands and silt. At present the task routinely determines or plans to determine Kd's for the following elements: Tc, Pu, Np, I, U, Cs, Ba (Ra substitute), Sr, Am, and a rare earth such as Eu or Ce for all groundwater-rock types studied. For a few rock-groundwater combinations data for the elements Sn, Cm, Sm, Yb, Rb, Ru and Ni have been collected.

About 2000 Kd values of a generic nature were produced which have been added to the data base from which preliminary rankings of geomedia with respect to geochemical acceptability will be performed.

The Kd data, groundwater compositions, rock and mineral characteristics, and nuclide concentrations have been installed into a computerized information retrieval system. This system will rapidly sort and select pertinent data from the large data base (approximately 7,000) to address the inquirer's special request. If a modeler needs Kd values for a particular scenario, the information-retrieval system will inquire about the type of scenario, and, after the operator inputs on the keyboard, the computer will select and display possible choices. The output from the system does not require a lot of technical interpretation.

Kd values were evaluated statistically as functions of rock, groundwater, and nuclide characteristics. This work will be a continuing effort as more data become available for analysis. Preliminary predictor equations for Kd's as functions of geologic and chemical conditions have been developed under a subcontract with Adaptronics, Inc. In FY 1979 Adaptronics performed the following statistical analyses on Kd data generated at PNL: correlation of adsorption and desorption Kd values per radionuclide per geomedia, predictor equations per nuclide over all geomedia and per media subgroupings, a comparison of model Kd predictions to actual Kd values from an independent subset, and rank-ordering of the importance of geologic-media characteristics and groundwater variables for modeling Kd per nuclide.

The significant results of this work warrant the conclusion that modeling Kd empirically as a function of geological and chemical characteristics can yielc very accurate empirical models. The parameters measured to describe the geologic media were shown to be significant in the modeling process, and the ability to establish their interactions has been demonstrated. It has also been shown that the perturbations in these interactions resulting in changes in Kd can be used to establish quantitative relationships which will facilitate sensitivity or parametric analysis.

Basic thermodynamic data were tabulated and incorporated into computer codes. The thermodynamic codes are used to evaluate the concentrations and species distribution that are expected at equilibrium. These codes will be used, in conjunction with literature data on natural reactors, natural ore bodies, and existing radionuclide disposal facilities, for validating short-time-span experimental studies with respect to long-time-span migration of radionuclides through the geosphere.

Most of laboratory sorption studies described above are concerned with generating needed data as rapidly as is possible without emphasizing the pursuit of a general understanding of the mechanisms controlling the adsorption-desorption.

More reliance on mechanism studies will be needed for extrapolation to situations not studied and for delineating the basic assumptions involved in migration simulation. Underlying assumptions are always involved in modeling efforts. Indeed, they frequently emerge as the key factors in assessing simulation results, such as done in a licensing process, rather than the output answers themselves. Also, extrapolation to geologic regimes not measured will inevitably occur, since all mineral combinations with all solutions under all physiocochemical conditions cannot be evaluated experimentally. For these reasons, mechanistic studies of the sorption/desorption processes will become essential for providing scientific defensibility for the assessments used for site qualification and licensing. Some work is under way to investigate the basic factors and their interactions that control the sorption/desorption processes of actinides, long-lived fission products, and activation products by geologic media. With a knowledge of mechanisms, migration-rate predictions can be performed from a more theoretical and fundamental basis for any proposed storage environment. The use of theoretical and fundamental data may reduce the costs expended to quantify the migration rates of nuclides at specific sites.

Efforts involved with this subtask include:

(1) Studies of multivalent radionuclides to look at sorptive and solubility properties as a function of valence state

(2) Investigations of nonequilibrium (kinetic) phenomena

(3) Use of classical ion-exchange theory to determine if observed patterns fit those predicted using data for Freundlich and Langmuir isotherms

(4) Use of microautoradiography and surface analysis at an atomic scale to identify the fate of radionuclides sorbed onto media

(5) Accelerated weathering experiments to assess the effect of weathering on sorption

(6) Studies of transport of colloids, clay-sorbed nuclides, and other nonsolution species.

Significant results to date show that ferrous iron-bearing rocks (such as basalt and many granites), some shales, and a few evaporites can reduce significant percentages of nuclides such as Tc, U, Np, and Pu (when they are present in trace, less than 10^{-7} M concentrations). The reduced species of these elements are characterized by very low mobility and by rather irreversible exchange properties. Preliminary results with fine colloidal kaolin particles show that some interaction mechanism (possibly electrostatic attraction) restricts the mobility of colloids in disaggregated rock columns. Thus the physical transport of colloids appears to be considerably slower than the groundwater and in fact may be insignificant.

Finally, in direct response to licensing activities, literature review and experimental studies were performed to obtain appropriate Kd values for safety assessments performed on site-specific exercises relevant to Paradox Basin bedded salt, WIPP bedded salt, and Hainesville dome salt. In addition to these studies, a report was prepared detailing the proper approach to be taken to obtain sufficient sorption-desorption data to license an actual site.

In FY 1980 all tasks previously described will continue, with data collection emphasizing environmental conditions germane to Gulf Coast salt domes and continental bedded salt. Extension of current experimental work, which considers far-field environments, to include near-field environments and total waste package-host rock interactions, is being proposed.

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RECENT RADIATION-EFFECT STUDIES ON NATURAL AND SYNTHETIC ROCK SALT FOR WASTE-DISPOSAL APPLICATIONS

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Extensive radiation damage will be formed in rock salt and other minerals adjacent to high-level radioactive waste canisters. If the canisters remain intact, only gamma-ray-induced damage will be formed. Appreciable additional damage will be created by alpha particles and beta rays if the canisters are breached and mixing occurs. If water is present, extensive mixing is expected. Radiation-damage information is essential for repository design, site selection, risk assessment, etc. To extrapolate from laboratory measurements to the long time periods and the dose rates that will be encountered in repositories, it is essential to develop a reasonably complete understanding of radiation effects in selected geological materials.

To date, the waste-disposal radiation-damage studies at Brookhaven National Laboratory have been concentrated on synthetic and natural rock salt. Studies are made with equipment for recording the optical absorption of samples, located in a temperature-controlled chamber containing inert gas, during and after irradiation with 0.5 to 3.0-MeV electrons. Spectra are recorded as often as every 40 seconds in the 3.1 to 1.55-eV (400 to 800 nm) region. This region contains the F-center band, which is a measure of the Cl^- ion vacancies, and the colloid band, which is a measure of the concentration and size of the colloid particles.

As shown by previous studies on natural and synthetic salt[1-3], during irradiation the F-band increases monotonically to a plateau while the colloid-band absorption remains at a low level until the F-center plateau is reached. At this dose the colloid formation rate changes from a small to a large value. Thus the colloid growth follows a nucleation and growth curve. This is shown in Figure 1. The growth curves are functions of dose, dose rate, temperature, strain, geological origin of natural samples, and numerous other factors. For example, strain applied prior to irradiation decreases the induction period (dose). Also, the colloid formation rate is low at 125 C, reaches a maximum near 175 C, and is negligible at 300 C. In several respects damage in synthetic NaCl is different from that in natural crystals.

Recently, radiation-induced colloid formation has been studied extensively. First, a computerized iterative procedure has been developed to separate the overlapping F-center and colloid bands. Second, the resulting colloid-band spectra have been fitted by a small-particle version of Mie theory to obtain information on the average diameter and the total amount of sodium in colloid particles. Proceeding on the basis that this procedure is reliable, one concludes that colloid formation in natural and synthetic rock salt is quite different. In natural salt the colloid concentration increases with dose but the colloid particle diameter remains nearly constant. In synthetic NaCl both the colloid concentration and particle size increase with dose. Other recent measurements show additional differences between natural and synthetic salt. Unirradiated synthetic NaCl contains an absorption band near 6 eV (210 nm) not present in unirradiated natural salt. This suggests numerous possibilities; e.g., impurities in one type of crystal may suppress or enhance radiation-damage processes occurring in the other type. The results have been used to evaluate the Jain-Lidiard[4] theory for radiation-induced colloid formation in NaCl. In many respects, the agreement between data and theory is quite satisfactory. However, the theory does not describe all of the observed effects. For example, the theory assumes that F-center diffusion is described by a single activation energy. The data for both synthetic and natural crystals provide one value at low temperatures and another at high temperatures. Most likely the theory can be modified to be very useful for repository calculations.

The research planned for the immediate future includes numerous studies on radiation damage in natural rock salt. Included are studies on strain applied before and during irradiation, studies on salt from different horizons in the same drill hole (probably ERDA-9) to determine if layer-to-layer variations are important, etc. Some measurements will be undertaken to improve basic understanding of experimental observations, e.g., the study of differences between natural and synthetic salt. Lastly, increased effort will be devoted to other rocks and minerals of interest to the waste-disposal program, particularly granite, basalt and shales.

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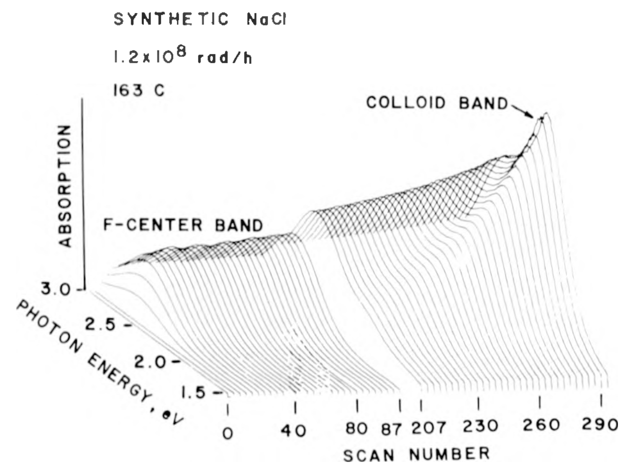

FIGURE 1. RADIATION-INDUCED ABSORPTION IN SYNTHETIC ROCK SALT MEASURED DURING IRRADIATION WITH 1.5-MeV ELECTRONS

Only selected spectra are shown. The F-band appears immediately and at the dose where it reaches saturation the colloid band begins to grow rapidly.

EXPECTED REPOSITORY ENVIRONMENT — SALT

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The objective of this study is to identify the expected environments for nuclear waste and spent-fuel repositories in bedded and in domal salt. The environments or effects include temperature distributions and histories, brine flows, brine chemistry, and pressure.

Thermal Environments

Since temperatures in the vicinity of the canisters depend on the separation between them, three-dimensional (3-D) models were developed to provide accurate prediction of the maximum salt temperatures.

The models have been applied to a variety of configurations and thermal loadings. For example, calculations have been performed for single-row configurations for both high-level waste (HLW) and spent fuel (SF). For HLW canisters loaded to 2.16 kw at emplacement and spaced in the row with an 8-ft pitch corresponding to 150 kw/acre, the maximum salt temperature occurs about 15 years after emplacement and reaches about 410 F (see Figure 1). For SF canisters containing a single 10-year-old PWR element separated by 5.12 ft in the row corresponding to 60 kw/acre, the maximum salt temperatures are predicted to reach 210 F after about 50 years. Reducing the HLW loading to 100 kw/acre and the SF loading to 40 kw/acre reduces the maximum temperature by 200 F and 40 F, respectively.

Two dimensional (r-z) models have been developed to predict the temperature rises which will occur within the emplacement hole and within the waste package. The results of these calculations in concert with the computed 3-D salt temperatures indicate that for HLW the maximum canister-wall temperatures will occur 5 years after emplacement and reach approximately 630 F for the design basis considered. For SF, the maximum canister temperature of 240 F occurs approximately 20 years after emplacement, representing about 50 F total temperature difference between the salt and the canister at this time. These calculations have assumed an annular gap between the canister and an overpack or sleeve and a 2-in. backfill of crushed salt having an effective conductivity of 0.1 Btu/hr-°F-ft, less than 5 percent of that for the contiguous rock salt.

A study of the dependence upon this effective conductivity and the thickness of the backfill region has been made to determine the possible effect of other backfill materials and engineered barriers. Predicted temperature rises for SF range from 200 F for 2 in. of backfill material having a conductivity of 0.025 Btu/hr-°F-ft to 50 F for a value of 0.1 Btu/hr-°F-ft.

Clearly, proper regard must be given to thermal conduction properties of materials chosen for engineered barriers surrounding the canisters.

Brine Migration

Brine trapped within the salt will migrate up the temperature gradient. A computer program, MIGRAIN, to predict this flow has been developed that utilizes the time-dependent continuity equation

$$\frac{\partial \rho}{\partial \tau} + \nabla \cdot (v\rho) = 0 \quad ,$$

which considers the resulting brine density (ρ) variations in the computation of the flow rates. The brine velocity (v) fields can be determined by an appropriate phenomenology, and computations have been made using the Jenks equation, which correlates experimental data for dependence of salt solubilities. The model agrees well with the experimental results of the Sandia Salt Block II experiment predicting the shape and, to within 30 percent, the magnitude of brine flow rates. The model does not predict the large flow rates occurring when sudden large decreases in temperature take place. This suggests that physical processes become important that are not simply described by the Jenks formula. The model has been applied to the case of brine migration in a repository for the heat-generating wastes. For salt containing 0.5 percent by volume of brine inclusions, the model predicts total brine flows of 15 liters per canister hole after 500 years for HLW and only 8 liters per hole after 1500 years for SF.

Fluid and Vapor Pressures

Once the temperatures and potential moisture conditions have been estimated, pressures in the vicinity of the canister can be predicted by invoking an appropriate equation of state and vapor-liquid equilibrium conditions. Utilizing an ideal gas flow approximation for the air in the emplacement hole, liquid and vapor fugacities to account for the coexisting phases, and the appropriate compressibility corrections, the total pressure in the emplacement hole can be estimated. In the simplest case, the emplacement hole is not air-tight, so increased pressure is due simply to the heated environment, and pressures are expected to remain below 30 psi. In the event the emplacement hole is sealed, pressures can exceed this value. For HLW, the maximum pressure may reach as high as 700 psi, assuming that closure of the emplacement hole due to salt movement takes place 100 years after emplacement. When the hole closes, due to creep, the canister will encounter the lithostatic and horizontal stresses associated with the depth of the repository. This is also true for SF, but prior to hole closure the total pressure is predicted to remain below 50 psi due to the lower temperatures in this case.

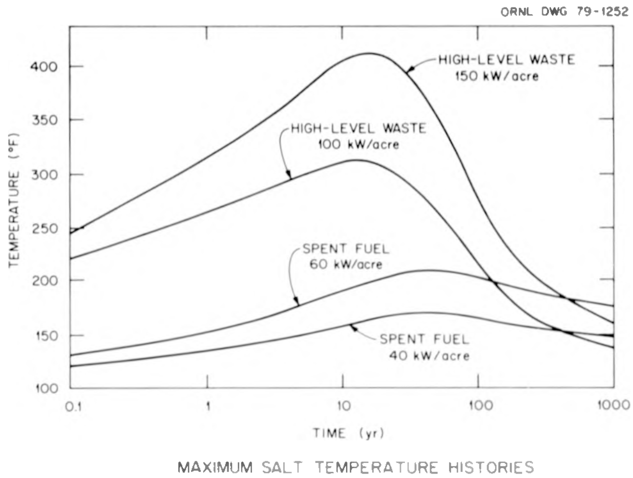

FIGURE 1. MAXIMUM SALT TEMPERATURE HISTORIES

STUDIES OF THE CONSOLIDATION OF SALT IN BRINE

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Oak Ridge National Laboratory

While the very presence of a subterranean salt deposit signifies isolation over geologic time from flowing water, somewhat ironically perhaps, the suitability of bedded salt for the storage of radioactive waste remains in question because the consequences of exposure to water have not been fully assessed. Water is inevitably present in small amounts as brine inclusions in bedded salt. Water may gain access to the waste repository during the operational period when it is open to the surface, or during the subsequent period of consolidation of backfilled salt, or, conceivably, even during the period of long-term storage. It is obviously necessary to predict in a salt repository the movement of brine in specified thermal and stress gradients, the effects of water on the mechanical properties of halite, and the effect of water on the rate of consolidation of backfill.

The purpose of our program is to study salt in brine under controlled conditions of thermal and mechanical stress in order to develop a model that can be used to predict behavior under the varied conditions of a waste repository. We report here our results thus far on the isothermal consolidation of salt under uniaxial stress in brine. We also describe a simple model being evolved to correlate and extrapolate these results.

Measurements of the consolidation of salt crystals in brine were made in an assembly consisting of a glass pipe (1.27 cm in inside diameter) fitted with a piston stressed by a Bimba air-actuated ram to compress the salt bed. The hydrostatic pressure drop across the salt bed was transmitted to a Validyne differential pressure sensor by means of brine-filled tubes. The height of the bed, the hydrostatic pressure drop across it, and the flow rate of brine through it were measured as a function of time.

The results of initial tests were not very reproducible, probably because fines were present in the milled salt used and because the salt crystals were not fully dispersed before the beds were formed. Fairly reproducible results were obtained using carefully sized, unmilled salt crystals which were fully dispersed in brine before the beds were formed. The results are represented approximately by the expression

$$\Phi = 0.827 - 0.0627 \log \left\{[\exp(0.046\,\sigma) - 1]\, t/z^2\right\}$$

when the void fraction Φ was in the range 0.4 to 0.2, the applied stress σ was varied from 20 to 100 bars, and the crystal size z was varied from 0.012 to 0.035

cm. The time t is in minutes. Because the exponential term was usually much greater than unity, a fair approximation to the data is given by

$$\Phi = 0.827 - 0.00125\,\sigma - 0.06727 \log (t/z^2).$$

Examination of samples removed from salt columns after consolidation usually revealed an increase in void fraction from top to bottom, presumably the result of a wall effect. As a consequence, the above expressions represent upper limits of the void fraction for a given applied stress, particle size, and time. We expect, however, when it is possible to make a proper correction for the wall effect, that the form of the above expressions will remain unchanged.

The permeability of the salt columns initially was close to expected values for the porosity and particle sizes involved. As consolidation proceeded, however, the permeability decreased with the fifth to sixth power of the void fraction, i.e., more rapidly than expected.

To correlate and extend the consolidation and permeability results being obtained, a model is concurrently under development which assumes the mechanism involved to be the enhanced solubility of salt crystals under stress. This pressure-solution effect is thought to be important in the consolidation of minerals to form sedimentary rock. The most elegant model appears to be that of Woyl[1] wherein the rate-determining step is assumed to be the diffusion of the excess dissolved material within the film of liquid between the stressed surfaces. This and a similar treatment of Kingery[2] led to the conclusion that for a given stress the void fraction should be a function of t/z^3. This does not appear consistent with our results.

For the present we have used a simpler model which assumes the rate step to be the diffusion of the supersaturated brine from the stressed grain boundaries into adjacent voids. This leads to the observed dependence of the void fraction of t/z^2. The dependence on the applied stress predicted by this model is also of the observed form, but the predicted numerical constant in the exponential term is much smaller than the observed value, suggesting that the actual area of contact between crystals under stress is only about 1 percent of the geometrical area. Moreover, this observed numerical constant does not decrease with compaction, as expected.

In future experiments we will (1) attempt to determine more accurately the form and magnitude of the wall effect, (2) determine the effect of applied stress on the permeability to brine of a boundary between two salt crystals, (3) study the effect of a temperature gradient, and (4) observe visually the processes that occur when salt is consolidated in brine. The further development of a model for salt consolidation, salt permeability, and water transport will proceed concurrently.

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REPOSITORY PERFORMANCE ASSESSMENT

WASTE-ISOLATION PERFORMANCE ASSESSMENT PROGRAM

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Office of Nuclear Waste Isolation

The Waste-Isolation Performance Assessment Program (WIPAP) is an ONWI-managed program to develop and apply modeling technology to evaluate the effectiveness of nuclear waste-isolation systems in preventing adverse radiological effects to living and future humans and their environments. WIPAP was established in October, 1979, by combining the far-field modeling activities at the Pacific Northwest

Laboratory and the far- and near-field tasks of the Performance Assessment Program procurement. To accomplish its intended purpose, WIPAP must organize present knowledge about isolation-system phenomena into a system of models describing possible radionuclide release scenarios and their consequences and use that technology to support the site-qualification, repository-design, engineered-barrier development, and system-licensing activities of the NWTS Program.

Waste-isolation performance assessment involves modeling the combined effect of system driving forces on system physical phases and predicting all phenomena significant to the system's performance. The scope of this modeling problem is depicted in Figure 1. The system is divided into many phases, not all of which are necessarily present in every isolation system. Because the system is bounded on the outside by the cosmos and on the inside by the Earth's core, Figure 1 is a picture of the universe as seen through waste-isolation performance assessment eyes. As shown, the isolation system can fail to perform its desired function as a result of driving forces working either from the outside in or from the inside out.

Two important characteristics of the assessment problem are apparent from examining Figure 1. The first of these is complexity. The system has many interacting phases and driving forces, and the assessment technology must analyze over positional scales from microns to hundreds of thousands of kilometers and time scales from microseconds to hundreds of thousands of years. The second characteristic is uncertainty. Because of limited understanding of some of the potentially relevant phenomena, limited accuracy of data, and limited ability to characterize the system without destroying its function, the assessment technology must analyze the uncertainty of the predicted results. Two more important characteristics are apparent from the anticipated assessment applications. The assessment results will support decisions which will be scrutinized in technical, legal, and political arenas. Therefore the auditability and comprehensiveness of the assessment logic are particularly important.

WIPAP approaches the problem in four tasks using the organization shown in Figure 2. Far-field scenario analysis defines the initial geosphere and biosphere states of the isolation system at the time of repository closure, simulates the effect of the cosmic, terrigenic, and human-caused phenomena and repository state changes on the system, and predicts future geosphere and biosphere states. Near-field scenario analysis defines the initial waste package and repository states of the isolation system, simulates the effect of the waste- and repository-caused phenomena and geosphere state changes on the system, and predicts future waste-package and repository states. Together, these tasks predict the development of transport pathways between the emplaced waste and humans and the initiation of radionuclide releases from the waste forms.

The near-field and far-field consequence analyses simulate the release and transport of radionuclides through the waste package, repository, geosphere, and biosphere and predict radiation doses to individuals and populations.

The WIPAP approach to scenario analysis involves a combination of classical systems, lumped parameter simulation, Monte Carlo simulation, and fault/event tree-analysis approaches. An integrated modeling system for far-field scenario analysis has been developed and applied to a specific site in a deterministic mode for univalued parameter distributions. Application in the stochastic mode for multivalued distributions remains to be done. Although many individual models for near-field phenomena have been developed, their integration into a modeling system is just beginning.

The WIPAP approach to consequence analysis involves deterministic modeling of groundwater flow, contaminant transport, water-dose, and air-dose. Development and rest application of an integrated modeling system for far-field consequence analysis involving multiple levels of model sophistication have been completed. Development of an integrated modeling system for near-field consequence analysis is just beginning.

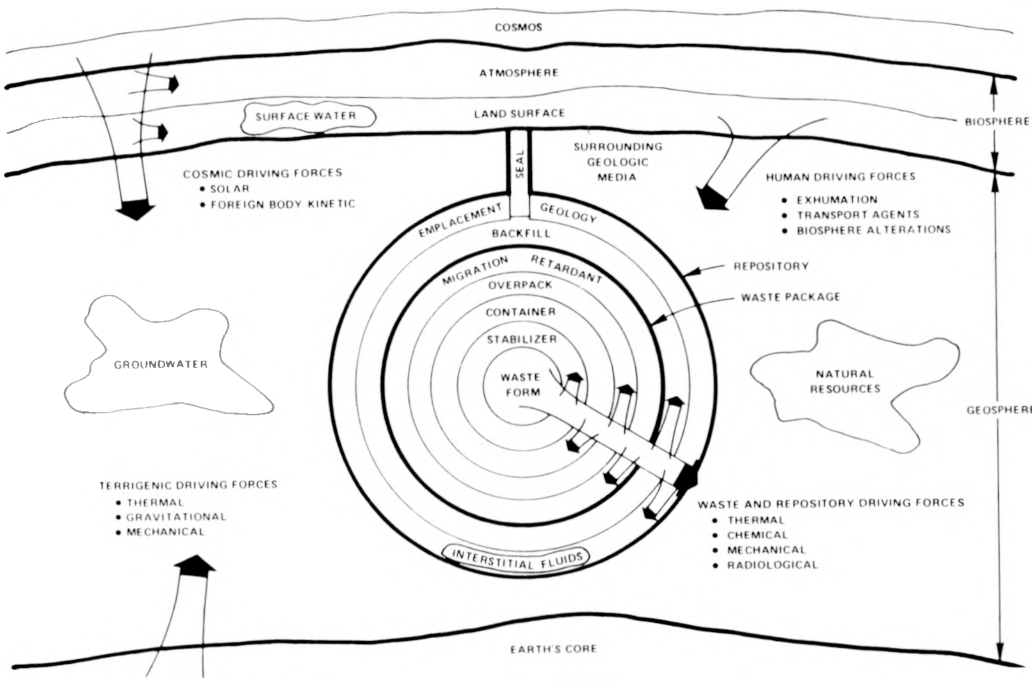

FIGURE 1. WASTE-ISOLATION PERFORMANCE ASSESSMENT PROBLEM

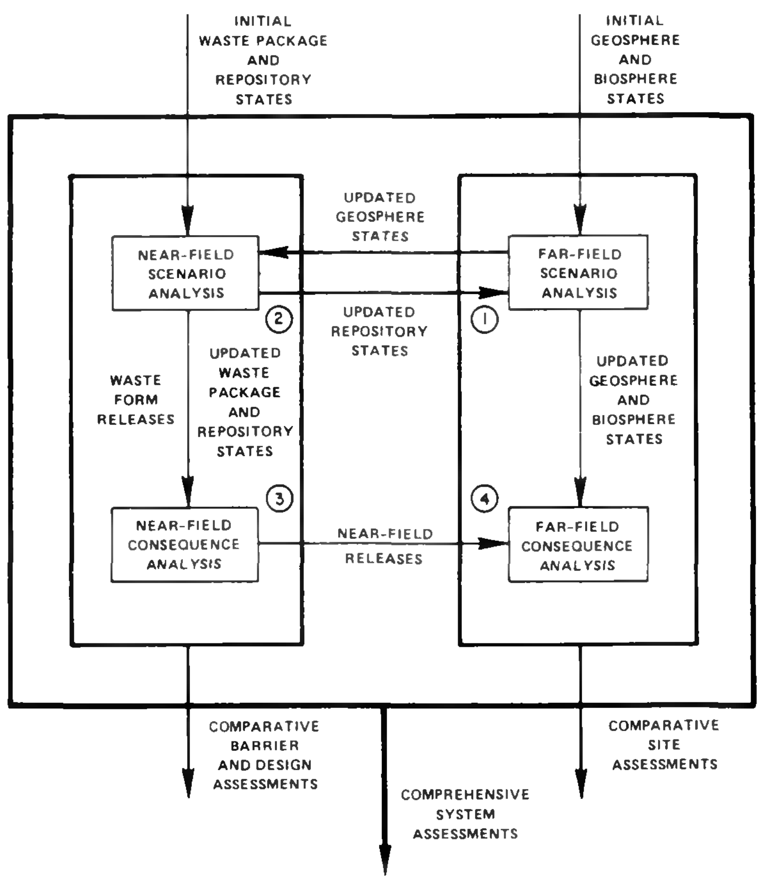

FIGURE 2. WASTE-ISOLATION PERFORMANCE ASSESSMENT PROGRAM (WIPAP) CONCEPTUAL ORGANIZATION OF ANALYSIS

RELEASE SCENARIO ANALYSIS TASK

G. L. Benson
Pacific Northwest Laboratories

The Release Scenario Analysis Task is being conducted as part of the development of a methodology to assess the long-term, postclosure safety of an underground nuclear waste repository. In essence, the long-term safety assessment can be segregated into two basic components:

(1) Identification and analyses of breach scenarios and the pattern of events and processes causing each breach

(2) Identification and analyses of the environmental consequences of radionuclide transport and interactions subsequent to a repository breach.

Within the PNL program (in 1979 and previously the WISAP program), the Release Scenario Analysis Task is charged with developing and implementing a methodology for assessing the first component. The objectives of the Release Scenario Analysis Task are:

(1) Develop and apply a methodology to analyze how disruptive phenomena may alone or in concert perturb a waste repository

(2) Identify the perturbations which result in a breach of waste repository

(3) Describe the state of the geosystem at the time of a breach as initial conditions for consequence analysis.

The current scope (FY 1979) of the program is such that it is addressing only the far-field parameters and is not at present addressing any near-field waste-induced phenomena. These important near-field phenomena are being addressed elsewhere in the NWTS/ONWI program and will be integrated with the Release Scenario Analysis Task methodology in subsequent efforts.

Any methodology developed for use in formulating disruptive release scenarios must address the problem within certain geological and hydrological system restraints. Some of these restraints are:

- Significant phenomena synergisms and event coupling
- Time dependence of geological and hydrological processes and events
- Limited data base for postulating future events and processes
- High degree of quantitative and qualitative system uncertainties.

Besides being capable of addressing in a positive fashion these system restraints, there are additional criteria imposed upon the methodology based on its ultimate use as a decision-making and licensing tool. Among these criteria are:

- Auditability
- Ability to accommodate objective and subjective opinions
- Facilitate parametric and sensitivity studies
- Facilitate or assist in describing breach frequencies
- Establish limits and/or initial conditions for input into the resulting consequence analysis after a breach is identified
- Be evolutionary and flexible in order to accommodate an increasing data base.

The repository simulation methodology developed to meet these many requirements is a brute-force, stochastic approach that incorporates Monte Carlo simulation, qualitative fault/event trees, and expert opinions. The repository simulation approach has six basic components: (1) characterization of individual disruptive phenomena, (2) identification of system synergisms, (3) characterization of Layered Earth Model (LEM), (4) development of system response rules, (5) simulation and Layered Earth Model evolution, and (6) characterization of potential breach scenarios.

The methodology is designed to work in two simulation modes: (1) user interactive and (2) a stochastic multiple simulation. When in the interactive mode, the user can pick the input parameters, select a set of system response rules, and highlight particular submodels for a single simulation run. In this fashion, the geoscientist users are able to utilize the model as both a stochastic and deterministic tool to facilitate understanding of the geological system surrounding the repository. This mode of operation will be particularly useful in developing a model that responds realistically prior to a full stochastic multiple simulation operation. The interactive mode will again be utilized to interrogate the breaches developed during the multiple simulation operation. In this manner, the sensitive components of the geological and hydrological system can be identified for additional parametric and sensitivity analysis.

The multiple simulation mode will allow a large number of independent simulations to be analyzed while choosing geological parameters from probability distributions and density functions. In this manner, the potential system response space which results from long-term interaction of the various geological phenomenon can be mapped. Within this response space potential repository breaches can be categorized for use as input into the consequence analysis hydrologic transport models.

The current status of the Release Scenario Analysis Task involves a second generation computer model that has been developed for a basalt repository. This model is currently being revised incorporating additional statistical capabilities. The models for salt domes and bedded-salt media are still in the conceptualization phase. Work during FY 1980 will be directed toward completing the basalt simulation model and developing initial simulation models for both salt media.

The methodology being developed by the Release Scenario Analysis Task is not being designed to "predict" the future geologic and hydrologic system. Instead, its objective is to attempt to bound, in a logical and auditable fashion, the set of potential future system states that could result in a repository breach. In addition to describing potential repository breaches, the output from this methodology will assist in describing the bounds for parametric and sensitivity analyses for the geohydrologic transport models. Thus, the consequence of potential repository breaches and the ultimate release of radionuclides to the biosphere can be analyzed.

RELEASE CONSEQUENCE ANALYSIS

J. R. Raymond and F. H. Dove
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This program has as its objective the development and application of methodology for assessing the far-field, long-term post-closure safety of deep geologic repositories. The work is being performed by PNL under contract with NWTS/ONWI. A specific task within the PNL program, developing a methodology for analysis of the consequences (water pathway) from loss of repository contaminant as defined by various release scenarios, is the subject of this paper.

Analysis of the long-term, far-field consequences of release scenarios requires the application of numerical codes which simulate the hydrologic systems, model the transport of released radionuclides through the hydrologic systems to the biosphere, and, where applicable, assess the radiological doses to humans.

Essentially three modeling technologies are involved in assessing the water-pathway release consequence. These models are: (1) hydrologic models that define the groundwater flow field and provide water flow paths and travel times, (2) transport models that describe the movement and concentrations of the radionuclides in the flow field, and (3) dose models that determine the resultant dose burdens to individuals and/or populations. Figure 1 is a schematic flow diagram for the release consequence analysis.

The various input parameters required in the analysis are compiled in a data system. The data are organized and prepared by various input subroutines for utilization by the hydraulic and transport codes. The hydrologic model simulates the groundwater flow systems and provides water flow directions, rates, and velocities as inputs to the transport model. Outputs from the transport models are basically graphs of radionuclide concentration in the ground-water plotted against time. After dilution in the receiving surface-water body (e.g., lake, river, bay), these data are the input source terms for the dose model, if dose assessments are required. The dose models calculate radiation dose to individuals and populations.

Hydrologic and transport models are available at several levels of complexity or sophistication. Model selection and use are determined by the quantity and quality of input data. Model development in the work provides three levels of hydrologic models, two levels of transport models, and one level of dose models (with several separate models). Table 1 lists the release consequence analysis models and their characteristics.

By the beginning of FY 1979, the PNL Release Consequence Analysis Task was at the point where site-specific test case analyses could be made:

- The data base was established on a flexible data retrieval system. Generic data were compiled for test-case model runs and verification. Data for a specific site were added to the system.
- The hydrologic models and transport models were implemented, sensitivity analyses had been performed, and test cases were run for verification.
- Previously developed radiological dose models were converted to the Univac 1100/44 computer system.
- A test case of the release consequence analysis methodology was completed for a reference site to exercise the Task models and their interfaces. Results indicated that the models could be utilized for initial assessments of site-specific cases, subject to availability of the necessary data and release scenarios.

During FY 1979, the following work was accomplished:

- A comparison release consequence analysis for the WIPP-site release Scenario 1 was initiated and completed.
- Sensitivity/parametric studies were completed for the reference Test Case Release Consequence Analysis.
- Release consequence analyses were initiated and completed for the Preliminary Information Report (PIR).
- Work continued on model and code modification, development, extension, and coupling. The extension of the FE3DGW hydrologic model to include energy and solute transport (including fluid density variations) was completed.
- Development work was initiated in several areas of technology describing fluid flow in porous and fractured media (inverse problem, dispersion, Kriging, and stochastics).
- Data-base development and operation continued for benefit of the test-case analyses.

For FY 1980, a major subtask will be the transfer of the release consequence analysis technology to the contractor(s) selected for the NWTS/ONWI Waste Isolation Performance Analysis Program (WIPAP). Work will also continue on performing additional release consequence analyses for specific sites and on model and technology development. The data-base system will be operated to support all of these activities.

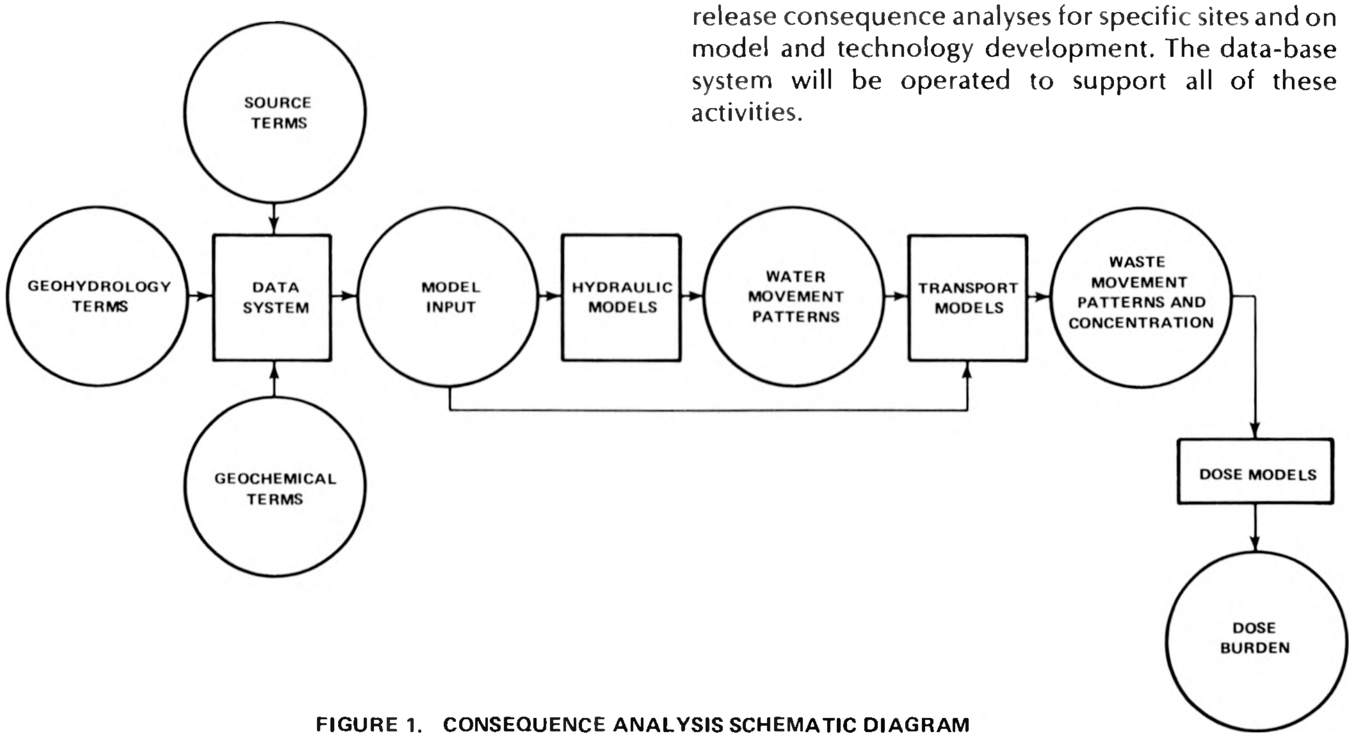

FIGURE 1. CONSEQUENCE ANALYSIS SCHEMATIC DIAGRAM

TABLE 1. CHARACTERISTICS OF THE RELEASE CONSEQUENCE ANALYSIS MODELS

Model	Characteristics
	HYDROLOGIC
PATHS	Two-dimensional, analytical/numerical methods, homogeneous geology, saturated flow
VTT	Two-dimensional multiaquifer, finite-difference numerical method, heterogeneous geology, saturated flow
FE3DGW	Three-dimensional, finite-element numerical method, heterogeneous geology, saturated flow
	CONTAMINANT TRANSPORT
GETOUT	One-dimensional, analytical method, chain decay, single speciation, equilibrium sorption, constant leach rate, dispersion
MMT	One-, two-, or three-dimensional, numerical method, chain decay, single speciation, equilibrium sorption, time-variant leach rate, dispersion
	WATER DOSE
ARRG	Drinking water, immersion, external shoreline, and aquatic food doses
FOOD	Terrestrial food dose
	AIR DOSE
KRONIC	Chronic external dose
SUBDOSA	Acute external dose
DACRIN	Chronic or acute inhalation dose

SESSION IIB
GEOLOGIC STUDIES IN THE NWTS/ONWI SITE IDENTIFICATION PROGRAM

DEVELOPMENT OF NWTS PROGRAM SITE-QUALIFICATION CRITERIA

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Office of Nuclear Waste Isolation

As part of its coordination role in the National Waste Terminal Storage Program, ONWI took the lead in developing DOE's site-qualification criteria for geologic repositories. The approach involved several stages, namely:

- Compilation and review of previous site-qualification criteria and creation of a draft document
- Review and comment by ONWI contractors familiar with site-qualification investigations
- Redrafting of the document
- Review and comment by NWTS Program participants and WIPP personnel
- Redrafting of the document
- Review by responsible DOE elements and NWTS Program participants to achieve program-wide concurrence.

Following concurrence of NWTS program participants, DOE will release the document for review and comment by interested members of the public.

The criteria are structured to address all site characteristics which affect waste isolation. Heretofore, emphasis was placed almost entirely on the host rock to provide isolation. However, repository licensing and eventual operation can proceed only when the total repository system can be shown to perform adequately to protect public health and safety. A systematic approach to site qualification, guided by predetermined criteria, is the first step in providing an acceptable geologic repository.

Ten generic areas of site-qualification criteria have been identified, as follows:

- Criterion I: Site geometry
- Criterion II: Tectonic environment
- Criterion III: Subsurface hydrology and geochemistry
- Criterion IV: Surface hydrology
- Criterion V: Geologic characteristics
- Criterion VI: Surface characteristics
- Criterion VII: Human intrusion
- Criterion VIII: Proximity to population centers
- Criterion IX: Environment
- Criterion X: Social, political, and economic factors.

These criteria constitute the framework of site-qualification requirements within which project-specific (i.e., site-specific) siting factors and specifications will be developed by NWTS Program participants. When such project-specific specifications are complete, they are to be issued as appendices to the DOE site-qualification criteria.

GEOLOGY AND HYDROGEOLOGIC MODELING IN THE SALINA BASIN, NEW YORK AND OHIO

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Stone & Webster Engineering Corporation

Geology

An extensive regional-reconnaissance study of the Salina basin has been made to evaluate the New York and Ohio portions of the basin for repository siting. An initial screening of the region was made to identify areas which warranted field study. The specifications used for the first-cut screening are discussed by Brunton and others.[1] Areas which contain an aggregate thickness of salt of at least 61 m, are underlain by salt at depths between 915 m and 305 m, and occur within regions having a seismic risk less than 3 (Algermissen, 1969)[2], were considered favorable.

Geologic information obtained from the literature and from public and private sources of unpublished data made it possible to evaluate these areas further.

A second screening was made based on data concerning faulting, oil and gas resources, salt mining, and brine-well locations. The specifications used were developed to allow concentration of effort during subsequent field programs in the most favorable parts of the study areas.

Recently, the specification for aggregate thickness of salt necessary for repository siting was decreased from 61 m to 21 m. The New York and Ohio regions were reevaluated to accommodate the reduced thickness requirement. New and significantly revised well data were employed in the reevaluation.

The results of the geologic screening are shown in Figure 1 for New York and Figure 2 for Ohio. The areas shown are considered favorable, from a geologic standpoint, for field study.

Results of the geologic studies also indicate that the geologic structure in the New York study area is complex. Folding and faulting within the salt section may make characterization of the salt geometry difficult. Ohio is less complex structurally, but has thinner salt deposits. Seismicity analysis shows that both study areas have had little earthquake activity.

Geohydrological Modeling

Two-dimensional salt-transport modeling techniques have been under development for use in the Salina basin investigations. Three modeling goals were established:

(1) To determine the relative importance (sensitivity) of the Salina basin geohydrologic parameters. This would aid in designing the field investigations.

(2) To provide a descriptive tool, for expressing the regional basin geohydrology, which could be continuously modified and upgraded as field data were collected.

(3) To aid in evaluating the long-term stability of the salt at specific locations within the basin considered for repository siting.

The modeling program has been based on the finite-element method and was developed by Dr. E. O. Frind of the University of Waterloo. The program was formulated in the vertical plane so that the effective driving forces, which are the hydraulic gradient acting on the boundaries and gravity acting on differences in fluid densities, could be properly represented. The program has not been qualified for Category I use.

Sensitivity analyses performed have concentrated on three salt-transport cases common to the Salina basin: (1) an aquifer directly overlying the salt, (2) an aquifer directly underlying the salt, and (3) an aquifer overlying the salt but separated from the salt by an aquitard (Figure 3). Results indicate that velocity (hydraulic gradient and permeability) and aquifer thickness are the important parameters for Cases (1) and (2), and velocity, aquifer thickness, aquitard permeability, and aquitard thickness are the important parameters for Case (3). Dispersivity plays a relatively minor role in all three cases.

Several regional profile models have been created through the New York study area, based on scanty data taken from the published literature and supplemented by well-log data from oil and gas explorations (Figures 4 and 5). Plots indicating salt concentrations, fluid pressures, and direction and velocity of fluid flow have been obtained from these models and appear realistic (Figures 6-8).

Further studies in this effort have been recessed pending resolution of Federal/State questions regarding the site screening program.

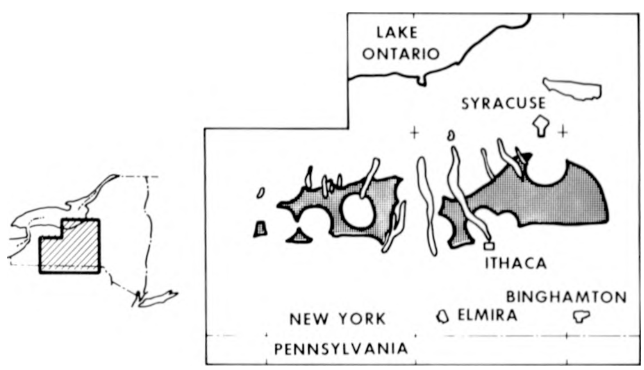

FIGURE 1. AREAS FAVORABLE FOR GEOLOGIC FIELD STUDY IN NEW YORK

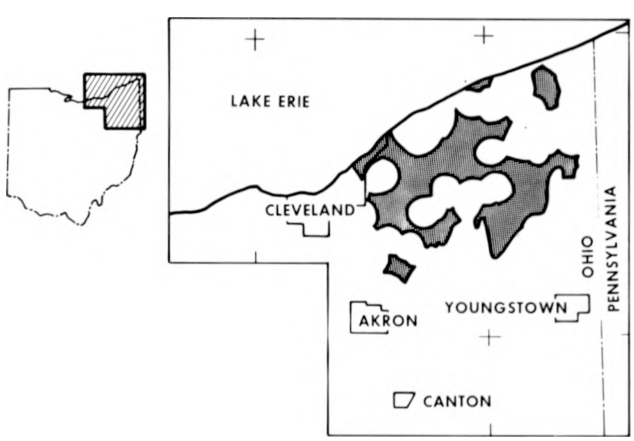

FIGURE 2. AREAS FAVORABLE FOR GEOLOGIC FIELD STUDY IN OHIO

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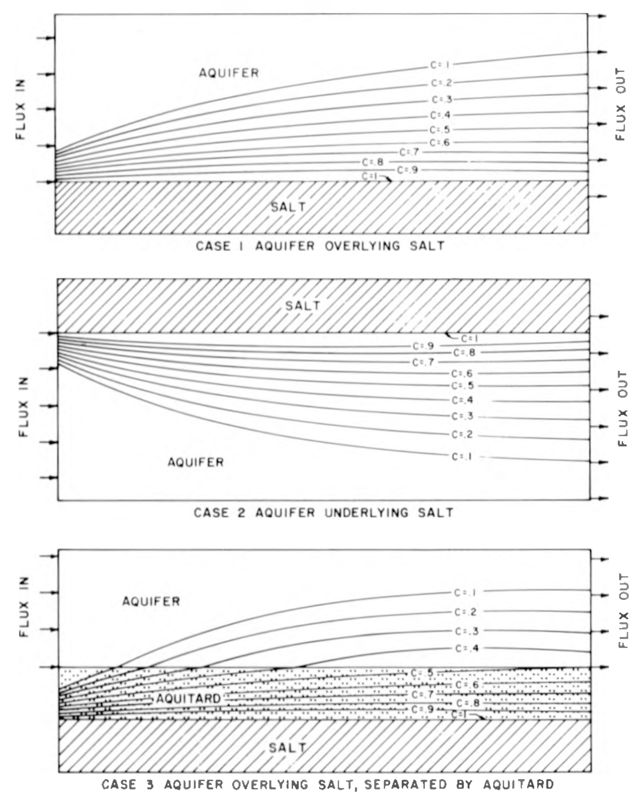

FIGURE 3. SALT DISSOLUTION, SALINA BASIN, NEW YORK

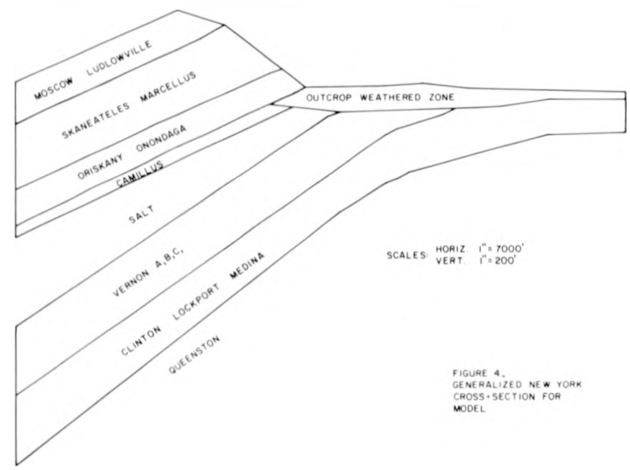

FIGURE 4. GENERALIZED NEW YORK CROSS SECTION FOR MODEL

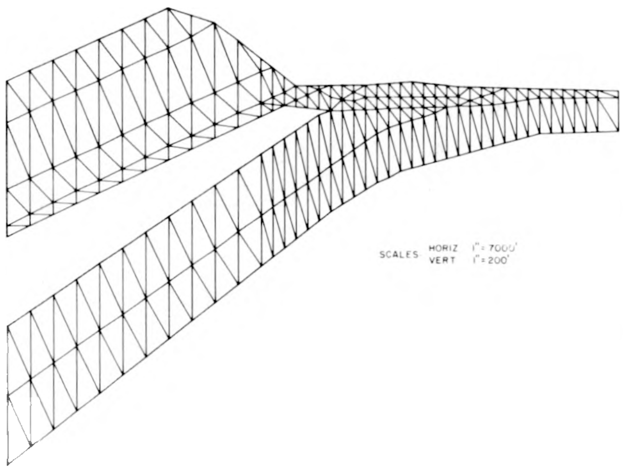

FIGURE 5. FINITE-ELEMENT MESH OF SECTION

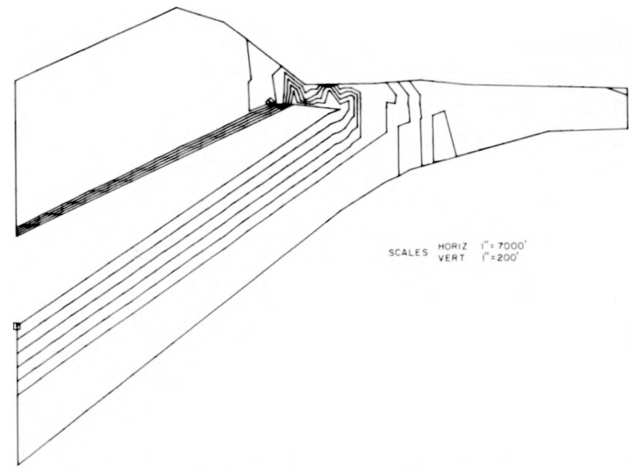

FIGURE 7. CONTOUR OF CONCENTRATIONS

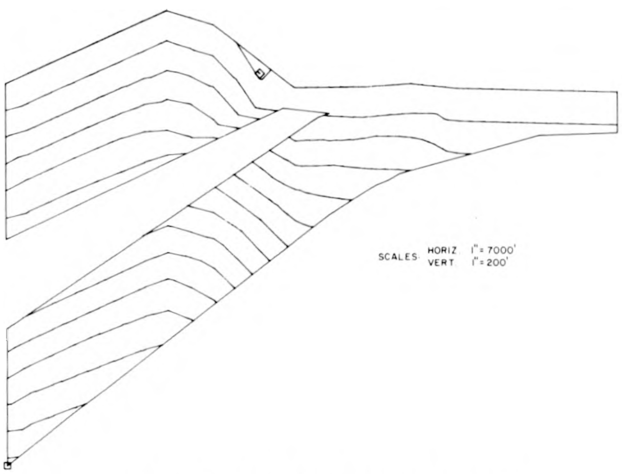

FIGURE 6. CONTOUR OF PRESSURES

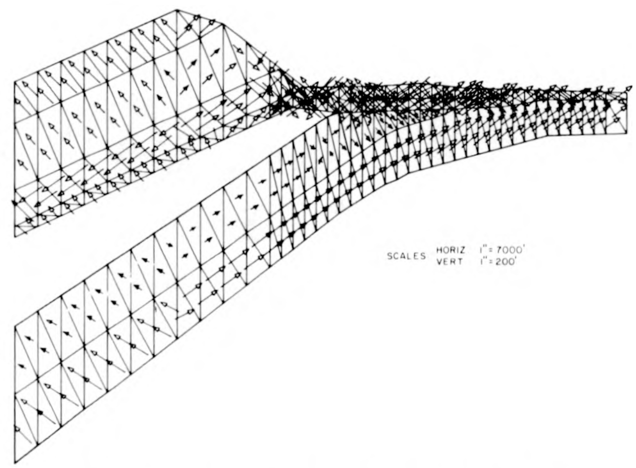

FIGURE 8. FLUX DIRECTION AND VELOCITY

SUMMARY OF STUDIES OF DEEP FORMATIONAL WATER ASSOCIATED WITH THE SALINA GROUP, OHIO AND NEW YORK

S. E. Norris
U.S. Geological Survey (Columbus, Ohio)

Studies of the deep formational waters associated with the Salina Group in Ohio and New York were part of a general investigation of the Silurian salt deposits in the Appalachian and Michigan geological basins to determine their usefulness as repositories for radioactive wastes. The studies were related specifically to the search for potential repository sites in northeastern Ohio and western New York. They were funded by the Department of Energy (DOE).

The first phase of the work, which was started late in 1977, consisted of a review of the literature for information on the occurrence, composition, and origin of natural brines. The literature review was completed in late 1978. A manuscript report, now being reviewed, presents information on the occurrence of brines in basin environments, their chemical properties with emphasis on Ohio brines, and a review of hypotheses relative to the origin of brines. The review is arranged chronologically, ending with the more modern concepts based chiefly on isotope studies. The bibliography developed from the literature search is listed at the end of this summary statement.

Most early hypotheses on brine origin stated that brines originated as connate or "fossil" sea water that was retained in the pore spaces of the rocks after the seas retreated and the rocks were elevated. The chemical composition of the brine was changed and its salinity was increased to up to 10 times that of ocean water. These changes were believed to have resulted from chemical reactions between the water and the sediments.

More recent work, based on studies of stable isotopes of hydrogen and oxygen, has established that some of the water molecules in brines originated as meteoric water. This has led some investigators to believe that the brines are principally meteoric in origin and that the connate waters have been largely flushed from the sediments in the course of geologic time. Others believe that the brines represent principally the original waters of deposition, modified by mixing with meteoric waters. Advocates of both views agree that brines owe their high salinity to an "ion sieving" mechanism, or selective concentration of ions by passage of water through shales. The shales have acted as semipermeable membranes that retained the salts while allowing some of the water molecules and selected solutes to pass.

The second phase of the work, which has been delayed, pending resolution of Federal/State questions on the site screening program, was to have been a field investigation of new commercial oil and gas wells, which were to be tested as they were being drilled. Plans had been made to measure heads, collect brine samples for analysis, and to determine the direction of fluid movement in the wells at various stages in the drilling. Preliminary arrangements had been made with drillers to obtain these data. Geophysical equipment, consisting of a logging truck and accessory components, had been assembled in the Ohio district office before the project was terminated. This phase had as its primary objective the collection of data upon which to base the choice of test drilling sites to determine the flow characteristics of deep-seated brines in areas being considered as possible repository sites. One such area, in northeastern Ohio, is the subject of a report by S. E. Norris (1979) entitled "The Silurian Salt Deposits in Eastern Lake, Northwestern Ashtabula, and Northeastern Geauga Counties, Ohio" (U.S. Geological Survey Open-File Rept. 79-269). The report gives the depths and thicknesses of five salt beds in a 250-square-mile area in northeastern Ohio in which further investigation, including test drilling, had been considered. Another report stemming from the studies is based on brine analyses collected in western New York by oil-and-gas drilling firms and state agencies. The report, now being reviewed, is entitled "Chemical Composition of Deep Groundwater (Brine) Within the Appalachian Basin, New York."

The brine analyses collected in New York are noteworthy in that the brines do not appear to increase in salinity with depth, as they do elsewhere in the Appalachian basin. This could indicate relatively rapid circulation of the deep-seated waters in western New York.

Important questions in brine genesis involve the forces required to move brines through the semipermeable shale membranes. One concept is that brines circulate through the rocks as components of regional flow systems, moving from potentiometrically high to potentiometrically low areas. Questions remain, however, as to the effects on such flow systems of the widespread deposits of bedded salt and anhydrite in the Michigan and Appalachian basins. Another concept concerning brine mobility involves osmotic forces which exist because of the presence of different salt concentrations on opposite sides of the shale membranes.

There are no plans to continue these studies beyond FY 1980.

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REGIONAL GEOLOGICAL SCREENING STUDIES FOR WASTE-REPOSITORY SITING IN THE PARADOX BASIN

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Introduction

These screening studies are part of the NWTS/ONWI site-identification effort in the approximately 19,300-km^2 Paradox Basin study region. The study objective is to identify and characterize a potential study area (in the size range of hundreds of square miles) that, based on available data, offers characteristics allowing the greatest ease of licensing from a geologic perspective.

Geologic screening studies were designed to achieve the above study objective and to be combined with concurrent environmental analyses (BNI, 1978; N. A. Norman and M. A. Glora, this volume, Session IIIb) in order to complete the regional-level analysis. Similar studies at more detailed levels will lead to an identification and comparison of potentially favorable candidate sites. The operational objective guiding this study sequence is to identify progressively smaller areas having progressively better-defined characteristics that are potentially favorable for siting.

Results of the regional geologic screening studies are shown in Figure 1.

Geologic Screening Process

Activities and evaluations comprising this process were performed in the following sequence: (1) preliminary selection of critical geologic factors that were judged to constitute potential assets or constraints to siting; (2) comparison of these factors with related or proposed siting criteria and selection of criteria; (3) gathering of data pertinent to critical factors and recording on regional basic data maps (scale 1:250,000); (4) evaluation of available basic data; (5) selection of screening factors and screening specifications; (6) preparation of screening maps; and (7) grouping and overlay combination of screening maps to identify and delineate potentially favorable areas. Recognizing that formal criteria do not yet exist and that current research may result in significant changes in the state of the art, the geologic screening process was designed with flexibility to accommodate changes in criteria or screening specification.

Critical factors, screening factors, and screening specifications

Critical factors considered as potential screening factors were as follows:

After evaluation of data pertinent to these factors in terms of completeness, location accuracy, suitability in addressing suggested criteria, and relative applicability at regional versus more detailed levels, the screening factors and screening specifications listed in Table 1 were selected.

Delineation and Utilization of Potentially Favorable Areas

Screening-factor maps were combined using an overlay method to identify and delineate areas having geologic conditions potentially favorable for repository siting (see Figure 1). The screening-factor maps were grouped into primary and secondary categories. Primary screening factors were the depth to the salt and the thickness of the salt. The combination of these factors resulted in delineation of several separate areas comprising the basic potential study area. These areas were divided into subareas having distinct characteristics by applying the secondary screening factors (see Figure 1).

Category	Factor
Repository-medium characteristics	• Depth to saline facies rock • Thickness and lateral extent of saline facies rock
Long-term geologic stability	• Faults • Degree of folding • Lineaments • Historical seismicity • Igneous features • Nontectonic deformation • Tectonic deformation rate • Erosion rate
Topography	• Degree of drainage incision
Economic resources	• Energy and mineral resources • Groundwater resources
Groundwater conditions	• Groundwater discharge zones • Complexity of groundwater conditions
Licensing considerations	• Traceable bedrock statigraphy • Quarternary features useful for age-dating most recent fault displacement, diapiric movement, or tectonic flexure • Absence of complex hydrogeologic conditions

These potential study areas were then combined with environmental screening maps in order to delineate recommended study areas for the Paradox Basin region.

Highlights of FY 1979 Results

During FY1979 the geologic screening study described above was begun and completed. This study was also integrated with a similar environmental study[1] in order to identify a recommended study area for the Paradox basin region.

Area-level geologic studies were initiated in those parts of the recommended study area in Utah having the potentially most favorable characteristics. Highest priority tasks were those addressing key licensing-related issues such as salt depth and thickness, fault capability, and hydrogeologic conditions.

General Plans for FY 1980

By the end of FY 1980, the present Paradox basin schedule calls for area-level studies to be nearly complete toward the goal of identifying a Most Favorable Potential Location. By mid-FY 1980, several potential locations are to be identified. Studies related to site identification will progressively focus toward a most favorable potential location (measuring tens of square miles in area). Topical studies addressing critical licensing issues such as fault capability and salt dissolution may continue elsewhere in and adjacent to the study region in support of site-identification efforts.

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TABLE 1. FACTORS AND SPECIFICATIONS USED IN GEOLOGIC SCREENING STUDIES IN PARADOX BASIN

Screening Factor (Concern)	Regional Screening Specifications
Depth to saline facies rock (isolation from biosphere)	305 m (1000 ft) to 915 m (3000 ft) - favorable 915 m (3000 ft) to 1,219 m (4000 ft) - potentially favorable
Thickness of saline facies rock (single salt bed 31 m thick)	Saline facies rock: thicker than 305 m - favorable 61 m (200 ft) to 305 m thick - potentially favorable less than 61 m thick - low probability of 31 m salt bed
Mapped faults (Geologic stability)	Areas within 8 km are potentially unfavorable
Igneous features (Geologic stability)	Areas within 8 km are potentially unfavorable
Groundwater discharge and groundwater use (isolation from biosphere)	Areas in immediate vicinity are potentially unfavorable
Energy and mineral resources (minimize potential for competing land use)	Areas with no identified conflict

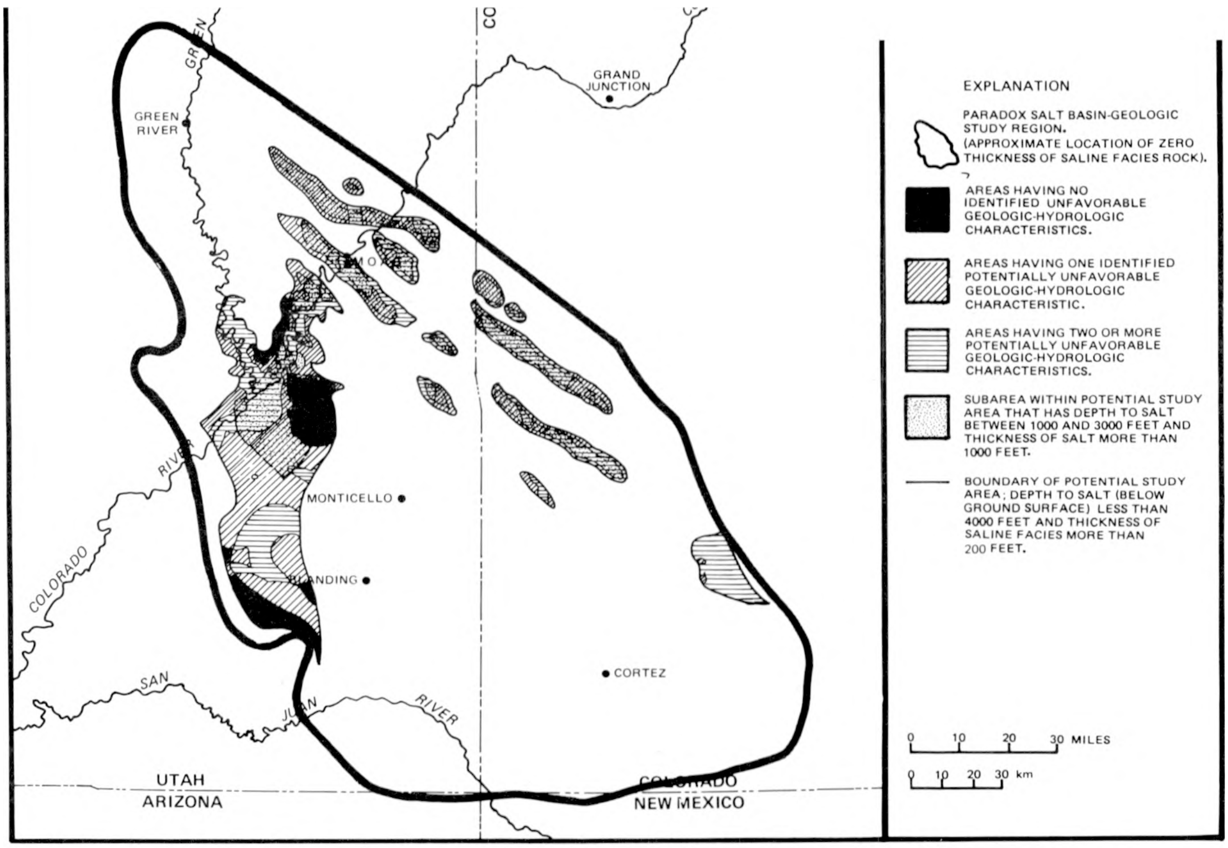

FIGURE 1. RESULTS OF REGIONAL GEOLOGIC SCREENING STUDIES OF THE PARADOX BASIN

GEOLOGIC EXPLORATION AT SALT VALLEY, UTAH

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Salt Valley is one of a series of northwest-trending diapiric salt anticlines along the northwest edge of the Paradox basin in southeast Utah. On the basis of a regional survey[1] of the salt deposits in the Paradox basin and two preliminary reports describing the subsurface[2] and surface geology[3] of Salt Valley, the Office of Waste Isolation decided that this structure should be investigated further. In 1978, a new investigation began under the Office of Nuclear Waste Isolation. This new work included a three-hole drilling program with geologic hydrologic, borehole geophysical, remote sensing, and surface geophysical work.

Geology

DOE No. 1 hole was drilled to a depth of 393.2 m (290 ft) and DOE No. 2 to 373 m (1227 ft). DOE No. 3 was cored almost continuously to a depth of 242 m (4074 ft). These holes penetrated the top of first halite at 167 to 196 m (548 to 644 ft). Using bromine geochemistry, borehole geophysical logs, and detailed lithologic logs of the core, we have been able to establish the structural configuration of the complexly deformed evaporite sequence intersected by the three holes. The water content of halite samples is being determined by a new technique of dissolving the halite in hot methanol and measuring the water content of the methanol extract by Karl Fisher titration. Analyses thus far indicate the average water content of the halite rock is about 0.21 weight percent.

Hydrology

At the drill site in Salt Valley, Utah, the unsaturated zone of the caprock is more than 100 m thick. The shallowest indication of water saturation was at a depth of 112 m (367 ft) in DOE No. 2, where water-table conditions prevail. Saturation was not identified in DOE No. 1 until a depth of 151 m (495 ft) was penetrated; subsequently, water rose in the well to a depth of 113 m (371 ft). Two components of potential groundwater flow were identified in the caprock, on the basis of apparent hydraulic gradients: (1) heads decrease as depth increases within the saturated caprock, causing a potential groundwater flow downward; and (2) head apparently decreases from DOE No. 1 southwestward towards DOE No. 2, resulting in an apparent component of flow in that direction. Groundwater in the caprock has ion concentrations that apparently vary within the saturated zone. Lowest specific conductances of samples were from the upper part of the zone, and were as low as 5,900 micro-ohms. Generally, deeper samples had much higher conductances. Two pumping tests of the saturated caprock yielded hydraulic conductivity values of 5 x 10^{-3} m/day and 4 x 10^{-3} m/day. These very low values would probably limit groundwater flow to very low rates. As a result, a long time would be required to circulate water through the caprock groundwater system. A ^{14}C age of a water sample from the caprock of greater than 36,000 years supports this conclusion. The $^{2}H/^{1}H$ and $^{18}C/^{16}C$ ratios indicated that the sampled water was derived from atmospheric water vapor by precipitation and infiltration. The $^{13}C/^{12}C$ ratio indicates that the carbon content of the sampled water had a biogenic origin; for example, it may be derived from petroleum. The salt and interbed sequence at the site had minor shows of oil and gas, but, in general, the lithologies have very low porosity and very low hydraulic conductivities. The latter generally are 1 x 10^{-4} m/day or less for five tests.

Geophysics

A vertical seismic profile (VSP) experiment was conducted in Salt Valley after the conclusion of drilling to determine if it is possible to map the position of the anhydrite-dolomite-black shale interbeds using this untried technique. With the initial effort, we obtained one complete fair-quality profile. Computer processing should allow eventual interpretation of the location and dip of interbeds. Two profiles in the immediate vicinity of the drill site were run with the "slingram" system, a loop-loop electromagnetic induction system that operates at five discrete frequencies. The penetration depth of the system is about 100 m. A slingram anomaly of about 200 m was sensed on both profiles and is likely due to a shallow hydrological feature. We also ran two vertical electric soundings (VES) of the Schlumberger type. Preliminary results indicate that there is no major conductor in the caprock section overlying the salt. Apparently, water does not have access to the top of the salt deposit at this location and has not formed a connected brine layer.

Borehole geophysical investigations, exclusive of the VSP experiment at Salt Valley, have been concentrated in the deep corehole (DOE NO. 3). A single coil induction probe gave good resolution of interbedded lithologies and may also be useful for determining moisture in halite rock. Studies[4] of the suite of geophysical well logs run at Salt Valley emphasize the difficulty of interpreting complex structural features.

A seismic refraction survey in the area of the drill site has revealed an undulating salt surface along the crest of the anticline that has local relief of as much as 80 m. The minimum depth to salt, about 165 m, occurs near the drill site.[5]

Remote Sensing

Lineaments of the northern Paradox basin in the vicinity of Salt Valley were mapped at 1:400,000 from high-quality Landsat images that were planimetrically rectified and processed for contrast enhancement and optimum display of fracture patterns and texture of the topographic surface.[6] Subsequent statistical lineament analyses and comparison of rosette diagrams reveal an impressive correspondence of strike frequencies of the lineaments and those of geologically mapped faults and fold axes, axes of gravity and magnetic-field anomalies, and zones of steepened gradient of the two potential fields. This analysis suggests that deep-sealed structures are reflexed through or have penetrated the overlying sequence of younger rocks, perhaps during late Phanerozoic reactivation of Precambrian basement structures. Structural control of major segments of the Colorado, Gunnison, and Dolores Rivers is also implied.

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A MULTIDISCIPLINARY GEOLOGIC APPROACH TO BASIN EVALUATION FOR NUCLEAR WASTE MANAGEMENT, PALO DURO BASIN, NORTHWEST TEXAS

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Nuclear waste-management studies of the Palo Duro and Dalhart basins in northwest Texas constitute an integrated, multidisciplinary approach to evaluating secure potential waste isolation sites in bedded Permian salts (Figures 1 and 2). Appraisal of all stratigraphic units in the basin, using cores, well cuttings, and logs, emphasizes genetic stratigraphy, facies, and depositional systems as a basis for interpreting basin hydrology and distribution of mineral and energy resources. Geomorphic studies of surface linear features, rates of erosion, stream incision and shallow subsurface salt solution are integrated with subsurface stratigraphy/structure, and hydrology. Hydrologic studies are designed to characterize deep basin circulation of fluids, and the hydrologic budget of the basin. The principal objectives are to evaluate salt appropriate for waste storage, to develop predictive models for judging lateral variations in salt quality, and to quantify long-term geologic and hydrologic processes which may affect the security of a repository.

Nonevaporitic sedimentary rocks below the salt-bearing section in the Palo Duro basin consist of 1000 m (3300 ft) of Pennsylvania and Lower Permian "deep" basin and slope shales, massive shelf-margin carbonates, and deltaic sandstones. Facies distributions define a major episode of basin subsidence and transgression (Pennsylvanian), with shelf-margin retreat, followed by regression, basin filling, and net southward shift of depositional environments (Permian). Early Permian carbonate shelf margins prograded rapidly seaward over slope and basin sediments and transformed the Palo Duro basin into an extensive platform with shallow water to supratidal environments on which salt, anhydrite/gypsum, dolomite, rare limestone, and red beds were deposited. Overlying the salt section are fluvial, deltaic, and lacustrine Triassic and Tertiary strata.

Evaporites and associated carbonates display basinward (southerly) facies changes from supratidal to subtidal; facies include: (1) salt formed on salt flats and in brine ponds, (2) bedded anhydrite, and (3) supratidal to subtidal dolomite. Red beds intertongue basinward into tidal sandflats. Clastic input was by eolian, alluvial, and/or tidal processes.

Calibration of well-log data with two continuous 1220-m (4000-ft) cores through the salt-bearing section will ultimately lead to predictive capacity for locating adequate salt lithology at a site-specific level. Massive-banded salt and chaotic mudstone-salt are identifiable on logs.

Permian bedded salts are undergoing in situ solution within 600 m (2000 ft) of the land surface. Coincidence of salt-solution zones and surface erosional features, including lake basins, major streams, and escarpments, suggests that surface erosion influences areas and rates of salt solution at depth. For example, collapse of sediments overlying salt-solution zones opens paths to the surface, permitting more rapid shallow groundwater circulation and, in turn, controlling patterns of surface erosion. Structure contour maps and stratigraphic cross sections of strata that overlie salt beds indicate that salt solution and subsidence of overlying beds began after deposition of Triassic formations.

Stratigraphic analysis of the deep basin sediments (pre-Triassic) resulted in identification of five potential aquifer systems, or hydrogeologic units: (1) pre-Pennsylvanian carbonates and minor sandstones, (2) Pennsylvanian-lower Permian arkose (granite wash), (3) Pennsylvanian-lower Permian shelf and shelf-margin carbonates, (4) Permian red beds, and (5) San Andres (upper Permian) carbonates.

From available drill-stem tests, pressure readings were converted to potentiometric head for water of variable density or salinity. In general, data are sparse but fairly reliable for lower Permian-Pennsylvanian carbonates and Pennsylvanian granite wash. Regional deep basinal groundwater flow generally is controlled by regional surface topography and basement structure. Flow paths are west to east and upward along the Amarillo Uplift.

The groundwater circulation system of the southern High Plains, eastern Caprock Escarpment, and Canadian river valley overlying the Palo Duro basin consists of (1) topographically high recharge areas, (2) extensive intrastratal flow paths (joints), and (3) regional discharge areas where flux is dominantly upwards (as expressed by high solute loads). Salt solution occurs today and accounts for the 2.7 x 10^9 metric tons per year (3 x 10^9 tons per year) solute load (Na, Ca, Cl, SO_4) of streams draining the southern High Plains.

Joints, which are probably major migration paths of groundwater in the High Plains aquifer, exhibit orientations similar to pre-Permian tectonic elements. Linear elements such as escarpments, stream segments, and aligned playa lakes parallel the joint orientations. Thus, it is likely that joint systems also control, in part, stream incision and propagation and slope retreat.

Slope, stream, and eolian processes and local climate are monitored at sites of maximum activity. Detailed field surveys of slope retreat, stream incision, sediment transport and deposition, and slopewash processes have been made in selected drainage basins. These data are related to total rainfall, rainfall intensity, available soil moisture, and freeze-thaw cycles to quantitatively characterize erosion rates under the present climate. Analysis of recent and Pleistocene climates offers a tool for extrapolating present data to obtain probable future rates of surface denudation.

Resources being evaluated include uranium, copper, evaporite minerals, water, and oil and gas.

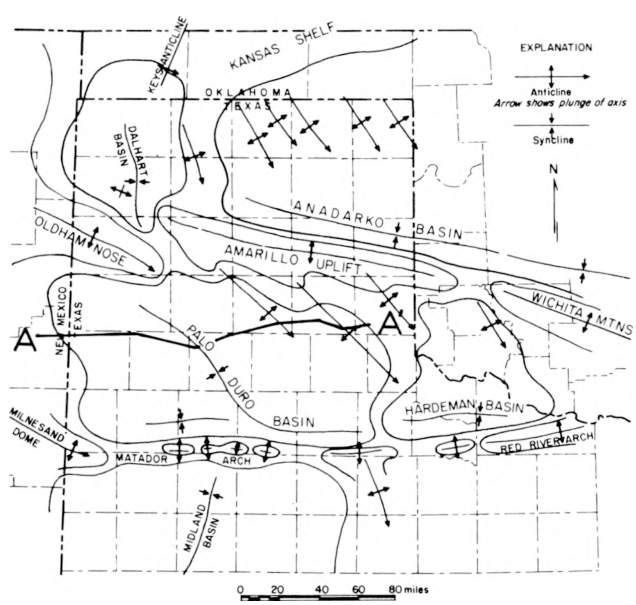

FIGURE 1. INDEX MAP OF STUDY AREA, SHOWING THE PALO DURO AND DALHART BASINS AND SURROUNDING AREAS

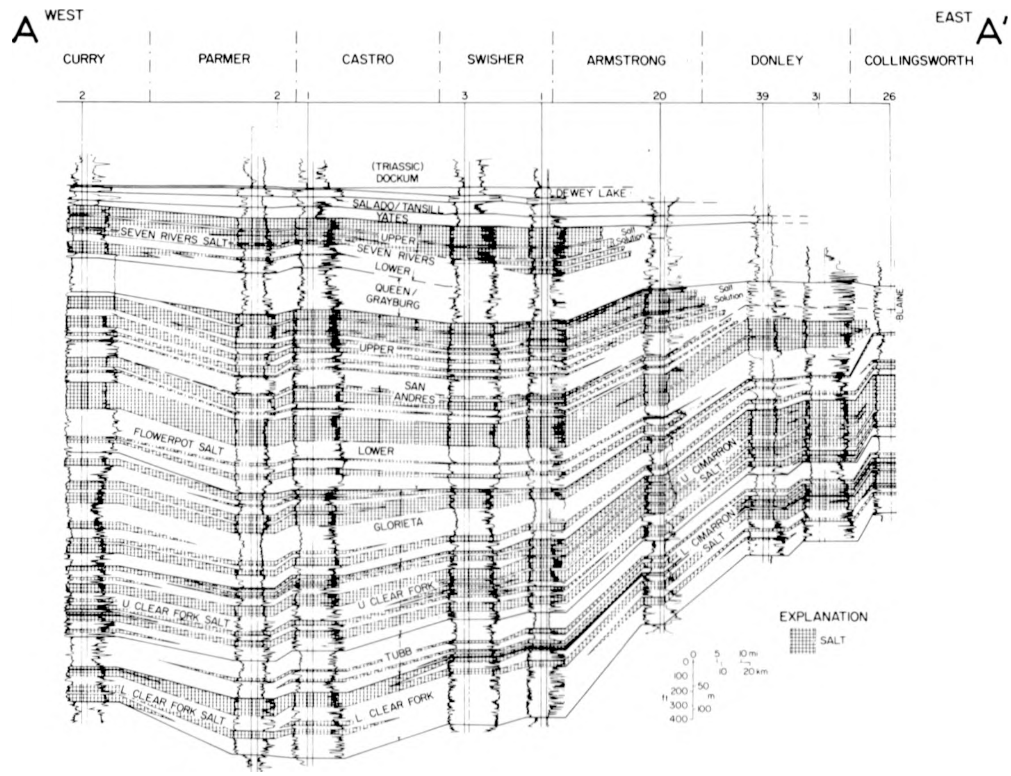

FIGURE 2. EAST-WEST CROSS-SECTION, UPPER PERMIAN SALT-BEARING STRATA, PALO DURO BASIN

GULF COAST SALT DOME PROJECT

A REVIEW OF GULF COAST SALT-DOME EVALUATIONS

C. O. Durham, Jr.
Law Engineering Testing Company

Many salt domes occur in the subsurface beneath portions of Texas, Louisiana, and Mississippi. These are roughly cylindrical salt monoliths extending vertically toward the surface from the parent salt bed at a minimum depth of 4 kilometers. Diameters generally range from 1 to 8 kilometers. There is considerable variation in individual domes, some having overhangs, others relatively tapered tops. Caprock of anhydrite-gypsum-calcite generally occurs above the salt. Coastal-plain sediments of Jurassic to Cenozoic age surround the domes.

Various domes have been exploited for petroleum from wells in adjacent and superjacent sediments, for salt from underground mines and solution cavities, for underground storage, and for gypsum and calcite in caprock quarries. Consequently, many pertinent data are available. In an initial evaluation of domes for radioactive waste storage, the 1975 United States Geological Survey report considered 36 of 263 domes as potential repository sites and emphasized suitability of interior domes in three basins in Northeast Texas, North Louisiana, and Mississippi, where earlier cessation of sedimentation has probably permitted dome stabilization.

Concurrent studies were inaugurated by the Office of Waste Isolation. Currently, the United States Geological Survey is conducting hydrologic studies. Louisiana State University, the Texas Bureau of Economic Geology, and the University of Southern Mississippi are conducting regional and dome-specific studies in their respective states. Law Engineering Testing Company is the geologic project manager, coordinating study activities and data acquisition. Law's initial task in 1977 was to assess the available data. By early 1978, eight candidate domes were recommended for preliminary evaluation. These were Vacherie and Rayburns in Louisiana; Richton, Cypress Creek, and Lampton in Mississippi, and Oakwood, Keechi, and Palestine in Texas. These were chosen primarily on the basis of size, depth, absence of mines or solution cavities, proximity of surface waters, and distance from large population centers.

Areas of 2500 to 3000 square kilometers (approximately 1000 square miles) have been defined in each basin to include the selected domes. Until early 1980, studies will characterize these areas and domes seeking answers to the following key geologic and hydrologic questions:

- What is the available space in the salt dome?
- Is the salt dome in a tectonically stable area?
- What is the lithology of the materials surrounding the upper section of the salt dome?
- What is the time length of ground water flow from salt dome to biosphere?
- Is there any significant mineral resource in the vicinity of the salt dome?
- Is there evidence of recent salt-dome development or movement?
- What are the effects of flooding in the area overlying the salt dome?
- Is there any significant unique feature of the salt dome?

Aquifers to depths of several thousand feet are being investigated in each basin. Individual wells screened in separate aquifers for pump tests are being drilled in clusters. Two to four clusters bracket each dome. In Louisiana and Mississippi these are supplemented by additional clusters dispersed through the area. Tests provide aquifer transmissivities, potentiometric surfaces, water quality, and other pertinent data.

Studies also include regional and dome-specific field mapping, with emphasis on Quaternary terraces, evaluation of remote sensing data and correlation of existing well logs and deep seismic data. Domes are being delineated better by gravity surveys, 6 miles of high resolution shallow seismic lines, twenty or more shallow borings, and one core hole drilled at least 500 feet into the salt.

Studies are designed to provide equivalent data on the three basins and eight domes. Activity has been hampered by varying access problems. Studies on one dome (Palestine in Texas) have been deferred because it has shallow dissolution problems associated with past brine operations.

Data acquired will provide the basis for selection of two or three domes for additional studies by March, 1980. These domes will be subjected to detailed site-selection studies lasting 1 year and designed to allow a thorough comparison of the domes regarding their design and licensing features. Work planned on these two or three domes includes additional hydrologic cluster wells, deep geologic borings, shallow geologic borings, high-resolution reflection seismic lines, two regional deep-reflection seismic lines crossing each dome, plus revised gravity modeling and other interpretations based on the new data. By March, 1981, these studies will have developed data adequate to permit comparison of the domes as potential repository sites.

Individual agencies are reporting on their studies in separate presentations that follow.

CYPRESS CREEK DOME, MISSISSIPPI

O. L. Paulson, Jr.
University of Southern Mississippi

The Cypress Creek salt-dome investigations are a part of the overall assessment of Gulf Coast salt domes as possible repositories for nuclear waste. The Cypress Creek dome is one of the three domes in Mississippi which passed the initial screening criteria for selection of salt-dome repositories.

The University of Southern Mississippi was selected to evaluate possible late Tertiary and/or Quaternary movement of the salt diapirs at New Augusta (Agnes or Cypress Creek) and Richton (Figure 1). These evaluations are being made from detailed surface geologic mapping. Domal growth has tilted and fractured surface formations in other parts of the Gulf Coast, and it was felt that similar effects on surface formations might be encountered during the present investigations at Cypress Creek and Richton.

Cypress Creek dome is located in Perry County, Mississippi, in the Mississippi salt basin, which is bounded on the south by the Wiggins uplift and on the north by the Pickens-Gilbertown-Pollard fault zone. The general configuration of the salt basin is shown in Figure 2.

The surface geologic units consist of the Miocene Hattiesburg formation and Pliocene Citronelle formation. In other parts of the Gulf Coastal Province, younger (Quaternary) terraces are located along many of the major stream courses. No such terraces were discernible in the Cypress Creek dome area.

Neither of the surface formations contains distinctive lithologies which can be correlated throughout the study area. The Pliocene Citronelle contains a clay unit which can be traced over a limited part of the study area, and the Miocene Hattiesburg formation contains both organic zones and bentonitic beds for which continuity is inconclusive on the basis of present data.

The surface mapping was supplemented with shallow borings by Law Engineering Testing Company. Gamma-ray logs were run on all but a few of the shallowest borings.

Limitations on (1) the number of exposures, (2) number and location of shallow borings, and (3) the lack of distinct lithologic units capable of being traced throughout the area leave a decided gap in the data needed to properly assess Cypress Creek dome. At the time of this writing, some final field checks and laboratory tests are being conducted, but we cannot say with confidence whether the surface formations have been affected by domal growth or not.

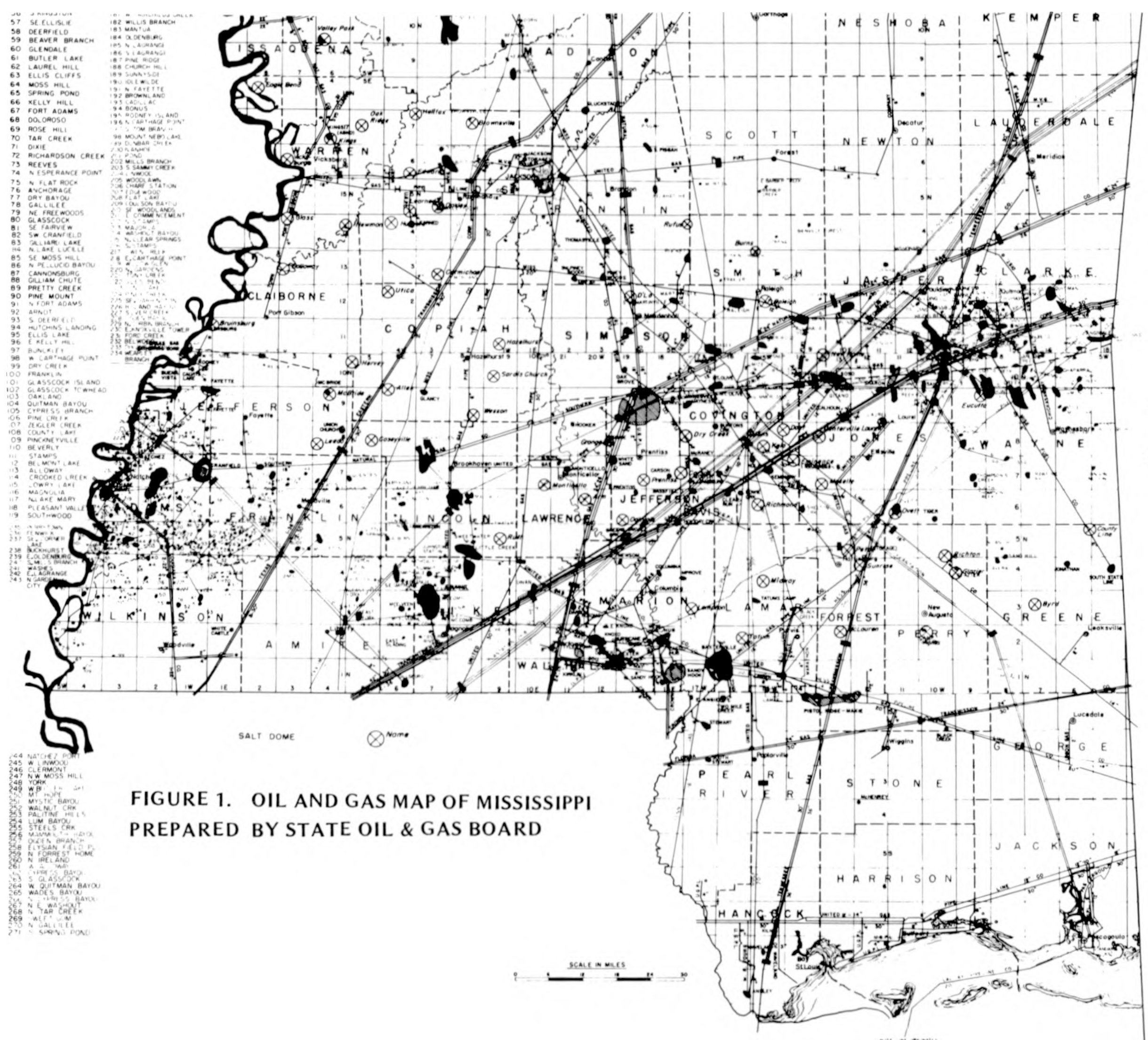

FIGURE 1. OIL AND GAS MAP OF MISSISSIPPI PREPARED BY STATE OIL & GAS BOARD

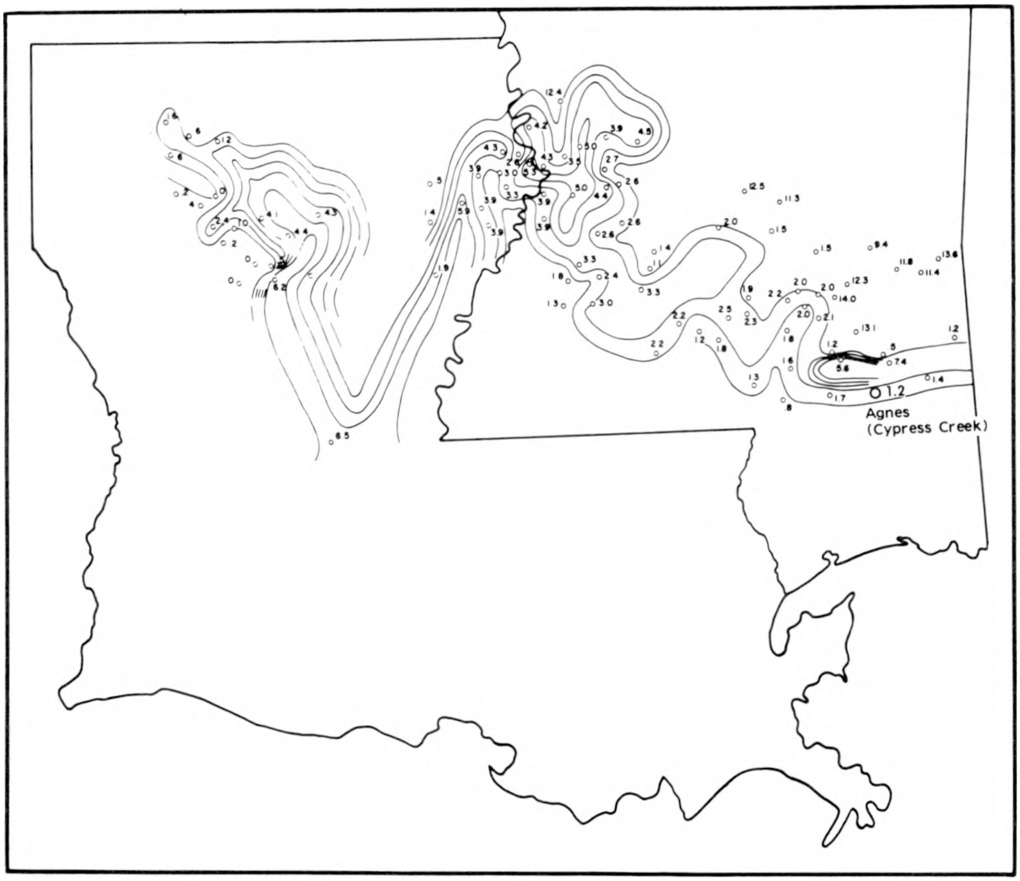

FIGURE 2. GENERAL CONFIGURATION OF THE MISSISSIPPI SALT BASIN

TECTONIC AND HYDROLOGIC STABILITY OF LOUISIANA SALT DOMES

J. D. Martinez
Louisiana State University

The Institute of Environmental Studies at LSU has just completed the fifth year of a program designed to evaluate the suitability of Louisiana salt domes for the storage or disposal of radioactive wastes. It has involved from the onset a multifaceted approach focused on the tectonic and hydrologic stability of salt domes. Although the site-specific studies have become limited to Louisiana domes because of programmatic needs, the scope and depth of the investigation have expanded materially from our initial involvement. A significant part of the LSU effort can be characterized as generic and, therefore, may be applicable to the entire region. In a more general sense, the LSU Institute for Environmental Studies supports the NWTS/ONWI program in the role of principal scientific advisor for the state of Louisiana.

Our effort to assess hydrologic stability can best be described as a determination of the resistance of the salt stock to external dissolution (Figure 1). A related hydrologic investigation initiated during the course of the work is the study of mine leaks in coastal salt domes. The purpose of this effort is to evaluate the degree of hydrologic isolation of a mined opening in a dome, by establishing the effectiveness of the salt stock to prevent inflow of water into and outflow of water from the mine (Figure 2). The question of hydrologic stability has been prinicipally addressed by the examination of logs from existing wells and the drilling of new water wells to establish the presence and pattern of salinity anomalies, "plumes" in fresh water aquifers penetrated by salt domes. The presence and thickness of caprock provides a long-term geologic basis for assessing stability, while the character of the caprock-salt interface furnishes evidence of present conditions. DOE's well-drilling program is currently hampered by lack of access.

In 1977 there was a major increase in LSU's effort with implementation of a contract to obtain a single deep core 2500 feet into salt in each of two north Louisiana domes. This operation had the necessary objective of providing exploratory probes into these domes to search for any data pertinent to their possible use as waste repositories. It also provided a unique opportunity to study petrology, geochemistry, and structure of two interior domes in single continuous cores. This can help in understanding the nature of the internal structure of these domes, which is essential to the prediction of the degree of hydrologic isolation provided by domal salt.

The question of tectonic stability has been approached in three ways on a temporal scale. Evidence provided by studies of Tertiary and older strata over 100 million years in age suggests that movement of the north Louisiana salt domes ceased perhaps as much as 30 million years ago. Later studies based on thinning of sediments over domes showed a slow and declining rate of growth through geologic time. Calculated Cenozoic values were well below 0.03 mm per year. This would amount to an upward growth of less than 15 feet in 150,000 years. These numerical values are very tentative and should be used with caution. Quaternary studies provide a perspective which extends through the last half-million years. Three Quaternary units believed to range in age from 25,000 to 500,000 years have been delineated at Vacherie dome. Careful study of the stratification of these units and their upper and lower boundaries shows no disturbance that could be attributed to domal growth. One long profile (Figure 3) encompassing Quaternary floored valleys to the north and south of the dome also shows no relative displacement above the dome.

Possible ongoing domal movement is being tracked by a system of instrumentation monitoring that is a part of the geomechanics effort. This includes the employment of tiltmeters, precise leveling, microseismic monitoring, and, possibly, laser ranging. Data obtained from the precise-releveling program (Figure 4) now indicate that the domes being evaluated are moving, if at all, only at very small rates near the threshold of instrument sensitivity detection. This technique has provided the most meaningful monitoring data obtained so far.

Work accomplished thus far has produced no conclusive evidence of either tectonic or hydrologic instability of Vacherie or Rayburns domes within the Gulf Interior Region. It is important, however, to call attention to the still-unresolved generic problem of hydrologic isolation (mine leaks). Progress is being made toward understanding and resolving this issue in Gulf Coastal domes.

Because of the unusual physical properties of salt as a rock type, a complete understanding of the genesis of salt domes in all details is difficult to achieve. LSU's research program has made us aware of certain features, such as mine leaks, sudden outbursts (or so-called "blowouts"), open cavities, and particular petrologic characteristics, which are poorly understood in coastal domes. Research is underway or is proposed to expand the knowledge in these areas. However, this potentially negative factor must be considered in the mix of data—much of which is positive—that will be evaluated in a decision-making process.

It has become increasingly obvious that these or any other problems associated with utilization of salt domes can best be approached by a consideration of the complete domal system (Figure 5). LSU's proposal for work in FY 1980 is in essence a continuation and extension of work described in this summary. The close linkage among the various components of our interdisciplinary team provides a good basis for the necessary systems approach required to resolve the issues identified in this review.

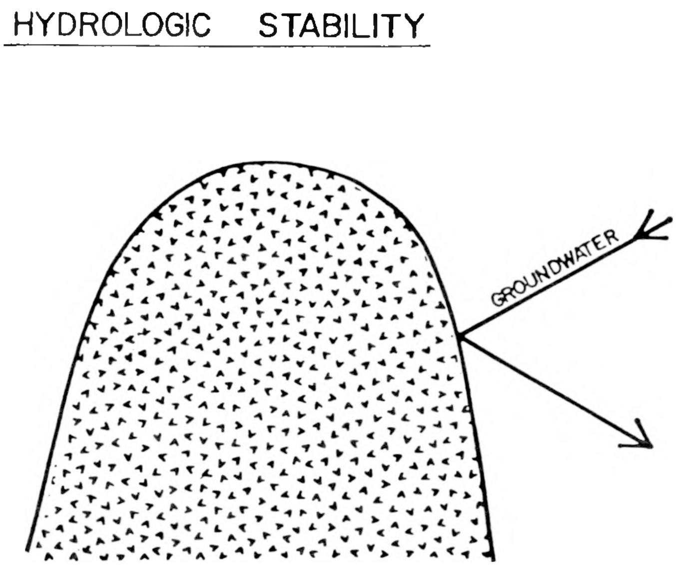

FIGURE 1. RESISTANCE OF THE SALT STOCK TO EXTERNAL DISSOLUTION

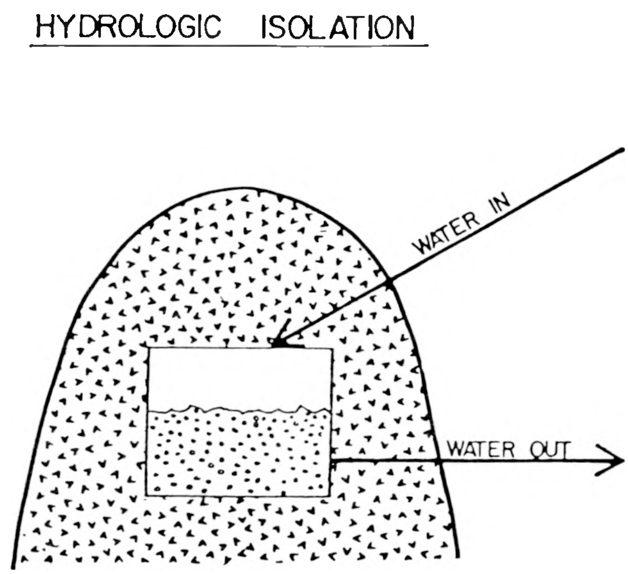

FIGURE 2. EFFECTIVENESS OF THE SALT STOCK TO BLOCK INFLOW INTO AND OUTFLOW OF WATER FROM A MINED OPENING

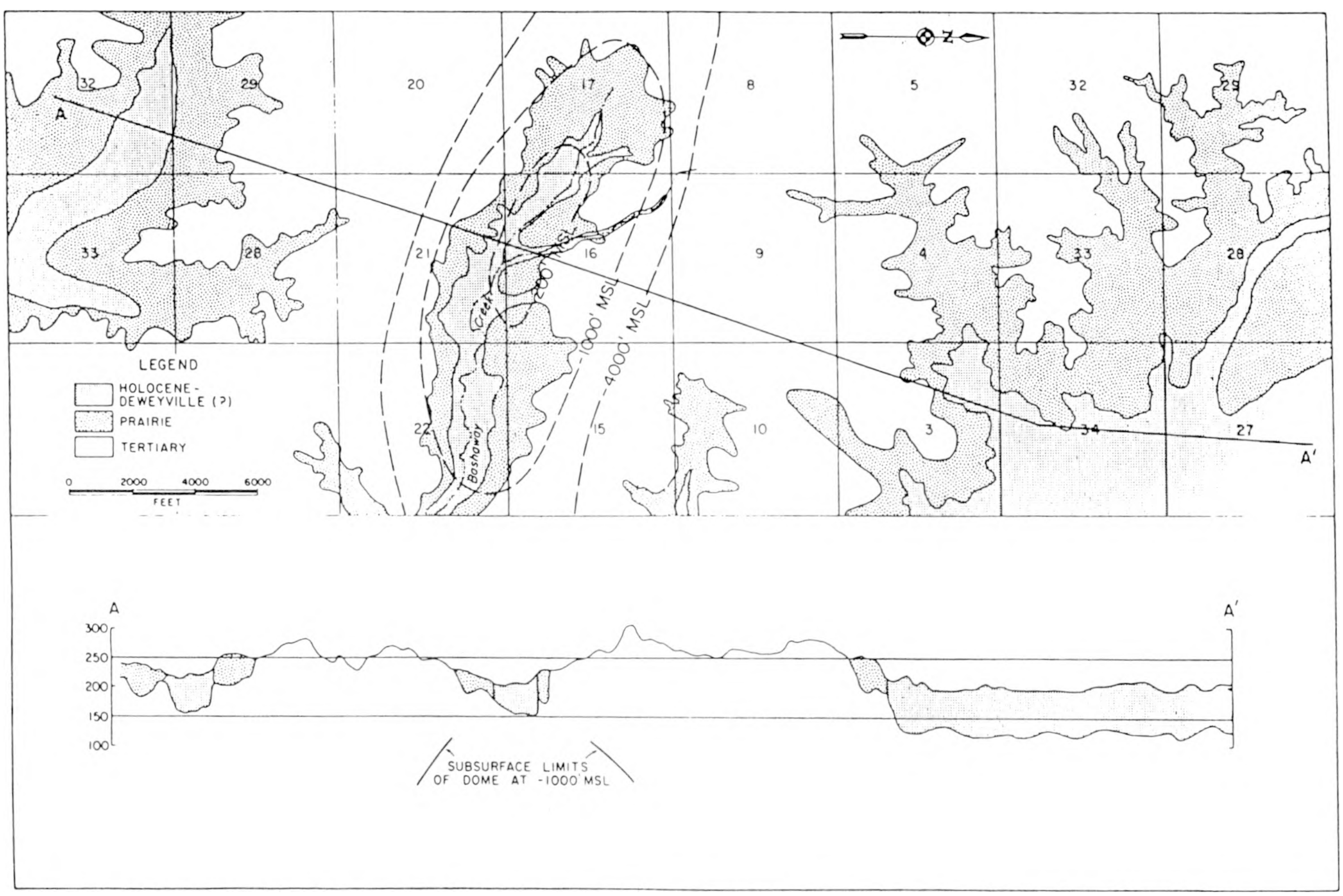

FIGURE 3. CONCORDANT LEVELS OF QUATERNARY DEPOSITS ALONG BASHAWAY CREEK AND SIMILAR DEPOSITS IN THE VALLEYS TO THE NORTH AND SOUTH

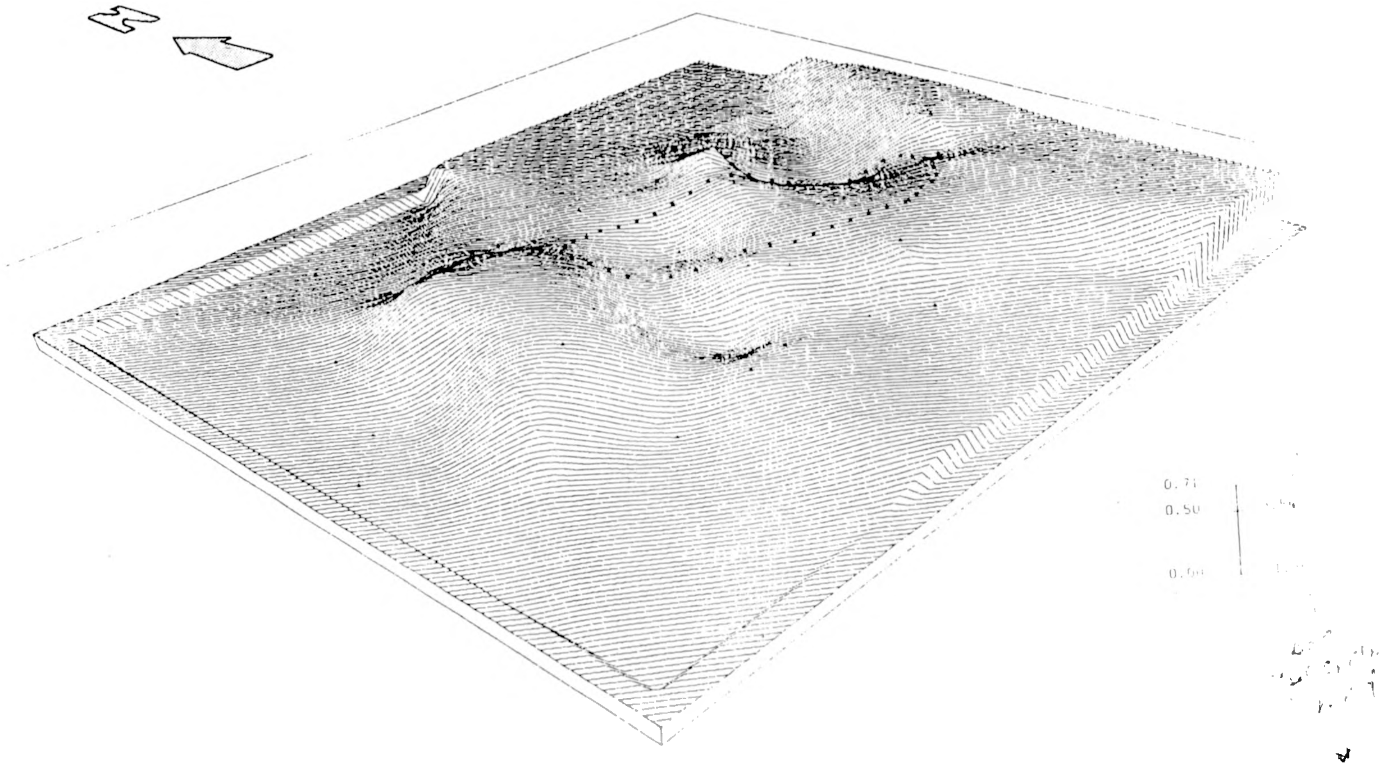

FIGURE 4. SURFACE ELEVATION CHANGE OVER VACHERIE DOME, VIEW FROM SOUTHWEST

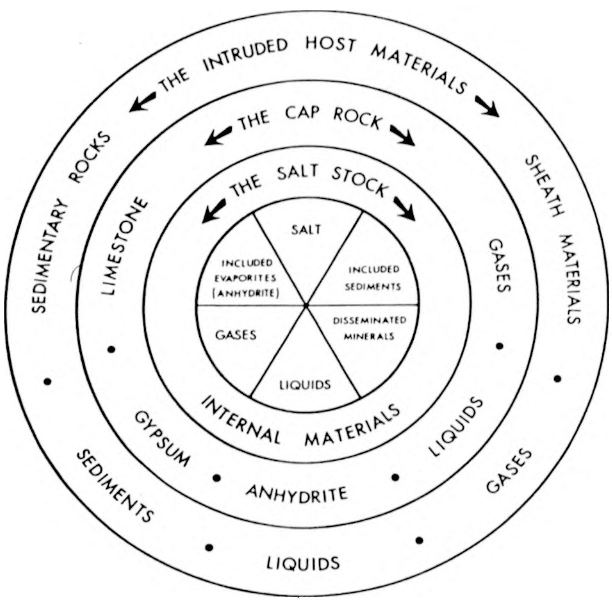

FIGURE 5. THE SALT-DOME SYSTEM

EVALUATION OF EAST TEXAS INTERIOR SALT DOMES

C. W. Kreitler
The University of Texas at Austin

The suitability of salt domes in the East Texas basin (Tyler basin), Texas, for long-term isolation of nuclear wastes is dependent upon hydrologic and tectonic stability of the domes and potential natural resources in the basin. These issues are being approached by integration of dome-specific and regional hydrologic, geologic, geomorphic, and remote-sensing investigations. Hydrologic studies are evaluating basinal hydrology and groundwater flow around the domes in order to determine the degree to which salt domes may be dissolving, their rates of solution, and the orientation of saline plumes in the fresh-water aquifers. Subsurface geologic studies are being conducted: (1) to determine the size and shape of specific salt domes, the geology of the strata immediately surrounding the domes, and the regional geology of the East Texas basin; (2) to understand the geologic history of dome growth and basin infilling; and (3) to evaluate potential natural resources. Geomorphic and surficial geology studies are determining whether there has been any dome growth or tectonic movement in the basin during the Quaternary. Remote-sensing studies are being conducted to determine (1) if dome uplift has altered regional lineation patterns in Quaternary sediments and (2) whether drainage density indicates Quaternary structural movement.

On the basis of the screening criteria of Brunton and others (1977), Oakwood, Palestine, and Keechi domes were chosen as possible candidate domes. Twenty-two domes have been eliminated because of insufficient size, too great a depth to salt, major hydrocarbon production, or previous use (such as liquid propane storage or salt mining or brining). Kreitler and others (1978)[2] tentatively concluded that uncertainties about three remaining salt domes, Keechi, Palestine, and Oakwood, needed to be resolved. The size of the Keechi salt dome was borderline; the Oakwood salt dome penetrated the Wilcox aquifer and also had oil production on an overhang; and a brining operation had been conducted in the Palestine salt dome during the early 1900s.

Since the work by Kreitler and others in 1978, the Palestine salt dome has been eliminated from further consideration as a potential repository because of the extensive deleterious effects from brining. Over 15 collapse sinks, resulting from the abandoned brine production, were discovered over the shallowest part of the dome. On the other hand, the Keechi salt dome has been found (by gravity modeling) to be large enough (1,810 acres) for a repository. The impact of the Oakwood oil field and the dissolution by groundwater of the Oakwood salt dome are not yet known, but are currently being investigated. Detailed geologic, hydrologic, and geomorphic investigations are now being conducted around the Oakwood and Keechi salt domes. They include the construction of four well clusters into the Wilcox-Carrizo aquifer around the Oakwood dome, and 20 shallow borings over the Oakwood dome to determine detailed surface geology over the dome.

Important conclusions are: (1) salt domes in the East Texas Basin are in different stages of dome growth when compared with the dome growth cycle of Trusheim (1960); (2) a surface fault in the Elkhart Graben probably has been active during the Quaternary, and the Mt. Enterprise fault system (to the east of the Elkhart Graben) appears active today; (3) the flow of groundwater around a salt dome and the process causing dome dissolution are very complex; and (4) the caprock over the Gyp Hill salt dome (South Texas) was formed by the accumulation of residual anhydrite sand left from the dissolution of the salt dome, and, even though much of the anhydrite section of the caprock is impermeable, the interface between the caprock and the salt has significant permeability and porosity.

Plans for FY 1980 include: (1) continued detailed geologic investigations (salt core, seismic studies) and hydrologic investigations (monitoring wells) of the Oakwood and Keechi Domes, (2) establishing seismic station(s) in the Mt. Enterprise-Elkhart Graben fault system, and (3) relevelling of N.G.S. benchmarks across the Mt. Enterprise fault system.

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(2) Krietler, C. W., Wermund, E. G., Guevara, E. H., Giles, A. B., and Fogg, G., 1978, Preliminary evaluation of salt domes in East Texas for National Waste Terminal Storage Program: U.S. Department of Energy, Interim Contract Report 16 p., 1 table, 9 figs.

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REGIONAL GROUNDWATER HYDROLOGY OF THE NORTHERN LOUISIANA SALT DOME BASIN IN RELATION TO THE STORAGE OF NUCLEAR WASTES

R. E. Hosman
U.S. Geological Survey (Baton Rouge)

The objective of the work of the Geological Survey in the northern Louisiana salt-dome basin is to determine the regional groundwater hydrology in relation to a general evaluation of the suitability of the basin's salt domes as possible repository sites for the storage of radioactive wastes. Transport by groundwater is considered to be the principal mechanism which might permit radioactive wastes to enter the biosphere. The work is supported by the Department of Energy as part of the NWTS Program.

The results of the first phase of this study are contained in the report "Geohydrology of the northern Louisiana salt-dome basin pertinent in the storage of radioactive wastes—a progress report" which was published as U.S. Geological Survey Water Resources Investigations Report No. 78-104. The report describes the geohydrologic framework of the system of multiple aquifers to depths of 3,000 feet or more in an area of more than 16,000 square miles. The Eocene Sparta Sand, which is penetrated by or overlies most of the salt domes, is the most productive and heavily used regional aquifer. Under natural conditions, groundwater movement is generally downdip from the outcrop areas of the aquifers. Discharge is to streams which intercept the aquifers or to overlying aquifers through confining beds.

Heavy withdrawals from an aquifer such as the Sparta Sand produce cones of depression in the potentiometric surface at the pumping centers, usually near municipalities. Water moves at an accelerated rate in areas within the influences of the cones. As pumping continues, the cones expand and can have a far-reaching effect on flow paths in the aquifer by changing the hydraulic gradient and directions of flow. Thus, a fluid entering the Sparta Sand, even though many miles from one of the pumping centers, would move toward this destination. The Sparta Sand is the only aquifer in the area for which sufficient potentiometric, hydraulic, and stratigraphic data are available to enable determination of regional flow paths. Plate 10 of the above-mentioned report shows the generalized flow paths and travel times for water from selected salt domes in the basin and indicates that movement of water is to the pumping centers. These results are preliminary and will be refined as additional data become available.

The regional flow conditions in and between the aquifers other than the Sparta Sand cannot be determined with the available data. Test drilling is planned to obtain hydrologic information for the major aquifers and confining beds. The objective is to develop a digital model that will simulate groundwater flow conditions in the basin. In preparation for the model development, 17 regional geohydrologic maps depicting geologic and hydrologic properties of all of the aquifers and confining beds have been completed. These maps are based on a compilation and analysis of all currently available data.

REGIONAL GROUNDWATER HYDROLOGY OF THE MISSISSIPPI SALT-DOME BASIN IN RELATION TO THE STORAGE OF NUCLEAR WASTES

C. A. Spiers
U.S. Geological Survey (Jackson)

The Geological Survey is evaluating the groundwater hydrology of the salt-dome basin of Mississippi with particular attention to the direction and rate of movement of water in the vicinity of the domes. The study is being made in behalf of the NWTS Program which includes a general evaluation of the suitability of salt domes as possible repository sites for the storage of radioactive wastes. The hydrologic characteristics are significant because transport by groundwater is considered to be the principal mechanism by which radioactive wastes might enter the biosphere.

There are 63 piercement-type salt domes in the basin. Penetration of the aquifers by salt bodies has produced a complex set of geohydrologic conditions. In general, little is known about the groundwater hydrology adjacent to and above the domes and few data are available anywhere in the basin about the deeper freshwater aquifers.

Three domes (Richton, Cypress Creek, and Lampton) have been selected for intensive study as possible sites for waste repositories. The geohydrologic study includes a comprehensive test drilling program near the domes. Most of the work in FY 1979 has been at the Cypress Creek dome.

Selected test wells are being completed as a network of observation wells to measure water levels and to sample the water for chemical analyses.

The oldest subsurface unit penetrated by oil-test drilling is the Louann salt of Jurassic age. The caps of most of the shallower piercement domes are in Tertiary strata ranging from Paleocene age (lower Wilcox) to Miocene age (Catahoula sandstone). These units consist principally of unconsolidated sand and clay with minor beds of marl, limestone, and sandstone.

The regional dip of the Tertiary units in the basin is south-southwest and it averages about 6 meters per kilometer. The principal structural feature affecting the shallower beds is the Perry basin.

Fresh groundwater occurs to depths of more than 900 meters in some places in the salt basin. The deepest freshwater strata are in the Wilcox Group in the northern part of the basin, in the Sparta sand and Cockfield formation in the central part, and in Oligocene and Miocene beds in the southern part.

Preliminary data indicate that the regional rate of groundwater flow in the Miocene aquifers ranges from about 30 meters per year to slightly more than 90 meters per year. The regional direction of groundwater movement is south to southwest. Deeper aquifers, generally less permeable and less affected by withdrawals and recharge, are probably characterized by lower rates of flow.

EVALUATIONS OF SUBREGIONS IN THE SOUTHEASTERN UNITED STATES

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Savannah River Laboratory

Geologic subregions in the southeastern United States are being evaluated to select areas for field study to identify potential sites for long-term storage of nuclear wastes. The southeastern United States contains several nonsalt rock types that may have a potential for siting a nuclear waste repository. Study of literature and existing knowledge of this region was begun by subcontractors in FY1978 to select broad areas between 100 and 1000 km^2 suitable for field exploration. Selections were based on geotechnical considerations only; socioeconomic and demographic criteria were not considered, although these criteria will have to be considered before a field study area is selected for investigation.

The region included the coastal states from Maryland to Georgia and was divided into three subregions based on geology: the Piedmont Province, consisting of crystalline metamorphic and igneous rocks; the Triassic basins, consisting of consolidated clay-rich sediments; and the Coastal Plain, consisting of unconsolidated to semiconsolidated clay-rich and sandy sediments. Each province was studied by a different subcontractor.

When a field study area is scheduled for investigation, initial geological and geophysical investigations from the surface would confirm that conditions indicated in the literature actually exist. Next, a subsurface exploratory program by means of boreholes would determine whether surface conditions could be extrapolated into the subsurface. From these extrapolations, one or more specific repository sites would be selected. Additional exploration would determine whether conditions at the specific site are the same as those of the region. The final step in site selection would be to excavate an exploratory shaft and tunnel that would adequately represent the site but not full-size. Hydrology and in situ rock characteristics would be tested in this tunnel.

General criteria for site selection were supplied to each contractor, who then translated them into a form that would be most useful for the type of rock in the assigned subregion. For example, the general criteria related to permeability state, "The geologic host formation should be of extremely low permeability and be surrounded by formations that permit no unacceptable leakage to the biosphere. These conditions should be simple and determinable." For the Piedmont study, this translated to "The area should be free of excessive jointing and should not be near any fault or shear zones"; for the Triassic study, "The area should not be intersected by numerous dikes or faults"; and for the Coastal Plain, "The sand/clay ratio should be less than 0.25". These translated criteria apply to the screening studies of literature and existing knowledge, where values for permeability or hydraulic conductivity are either generally not available or of unknown applicability to the area and the depth of interest. More specific criteria could be given for application to the results of field studies where the information developed was more specific.

This paper summarizes studies in the three subregions. In FY1980, an executive summary will be prepared.

Piedmont Province

The Piedmont Province consists of metamorphic and igneous rocks that range in age from the pre-Cambrian to late Paleozoic. There have been several episodes of metamorphism during this time. The late intrusive plutons appear to be the most favorable field study areas because the rock is more homogeneous than the surrounding metamorphic rocks; thus, the results of exploration are easier to extrapolate. Because of this homogeneity, the hydrogeology is expected to be easier to explore, and fractures may be less than in the more foliated and schistose metamorphic rocks. The structural stability of the rock would be greater than that of the phyllites and argillites. Thirteen plutons in the southern Piedmont are large enough to be considered field study areas. However, socioeconomic criteria would reduce this number.

Because the Piedmont is complex, the literature was surveyed in more detail in FY1979 by four university professors to ensure that the metamorphic rocks had been adequately assessed.

Triassic Basins

The Triassic basins along the east coast of North America are long, narrow troughs of sedimentary rocks in the metamorphic rocks of the Piedmont Province; some are buried beneath the Coastal Plain. These filled basins are commonly several thousand meters thick and consist of mudstone, shale, and sandstone of alluvial and lacustrine origin. Some of these basins also include coal, basalt flows, and freshwater limestones. Diabase dikes are also common.

The basins commonly have a faulted border on one or both sides. In addition, some basins have cross faults and later longitudinal faults. The frequency of dikes and faults was considered a significant criterion because these features may be avenues of groundwater flow.

Of the 19 known exposed and buried Triassic basins in this region, 6 had sufficiently large amounts of argillaceous rock and were sufficiently free of dikes and faults to be considered as candidate field study areas based on geotechnical criteria.

Coastal Plain

The Coastal Plain consists of a wedge of unconsolidated and semiconsolidated sands, clays, and limestones that cover Piedmont-type rocks. The principal applicable criteria for field study areas in the Coastal Plain are lithology, hydrology, depth of the host rock, and properties of the rock surrounding the host rock. Four areas fulfilled the applicable criteria.

GEOLOGIC EVALUATION OF CRYSTALLINE INTRUSIVES AND SELECTION OF CANDIDATE AREAS FOR DETAILED INVESTIGATIONS

G. W. Murrie and T. M. Gates
Dames & Moore

A geologic evaluation (based on available literature) of crystalline intrusive rocks in the U.S. was conducted to determine their distribution and potential suitability as repository hosts. The geologic characterization of these deposits relevant to nuclear waste isolation was performed and included such considerations as tectonics, seismicity, hydrology, mineral resources, and physical-chemical properties.

The objective of placing radioactive wastes in deep geologic host media is to isolate radionuclides from the biosphere. Previous RD&D activities have concentrated on salt, but recently initiated programs include alternative media, such as crystalline intrusive rocks. This group includes medium-to coarse-grained igneous intrusives and high-grade metamorphics with similar properties (e.g., gneisses, migmatites, etc.). These rocks form extensive, widespread deposits (Figure 1) which are relatively homogenous and exhibit high mechanical strengths, low porosities, and low permeabilities.

General geologic criteria utilized for this study are listed in Table 1. One of the criteria is hydrology. Permeability in unfractured cyrstalline rock decreases with depth, and large volumes with permeabilities of 10^{-7} cm/sec and less have been found.[1] Evaluation of regional and local hydrology can be based on rock-mass permeabilities, hydraulic gradients, groundwater-flow patterns, and relationships to aquifers and drainage basins.

Faults exert a deleterious effect on candidate host rocks because they provide avenues of permeability for water and affect mining operations and the structural stability of the repository. Geologic and tectonic maps were supplemented with lineament traces from Landsat photos and published descriptions of metamorphism and jointing to evaluate individual areas.

Regions with low seismic risk are favored in repository siting. Areas lying in seismic-risk Zone 3 (major damage) have been classified as unacceptable, although studies indicate considerably less damage to underground openings than to surface facilities immediately above, indicating pronounced attenuation with depth.[2]

Locations in and near regions which have experienced volcanic activity in the last 1-1/2 to 2 million years are unacceptable as repository sites. Quaternary volcanics within the U.S. are confined to the west, and occupy an estimated 65,000 km^2 of potential volcanic terrain.[2]

Tectonic stability, flooding, and glacial risk have also been considered in assessing suitability of different deposits. The total amount and rates of uplift and subsidence in the U.S. over the last 10 million years have been compiled[3], and areas located in highly unstable regions were eliminated from consideration. Flooding risk was evaluated for low-lying areas and closed basins. Quaternary sea-level oscillations have developed terraces that now stand at elevations up to 140 m above present sea level.[4] Glacial risk was also evaluated, although it seems to have a lower impact because relatively large blocks of undisturbed Precambrian shield rocks in Canada have undergone numerous glaciations without major fracturing.

Denudation rates for the U.S. vary between 40 and 165 m/million years.[4] In the case of a repository situated at the minimum depth of 305 m, breaching through erosion would be expected at 6 to 25 million years, assuming similar rates of erosion. This far exceeds the proposed target life of a repository.

Other criteria which influence site selection are the thickness, depth, and lateral extent of deposits and the host-rock properties. Consideration was also made of the possibility of natural-resource conflicts in areas of known mineral and energy resources.

Information gathered in this study indicates that numerous regions in crystalline intrusives in the U.S. are potentially suitable candidates for further investigations. Two matrices were prepared utilizing the above listed criteria to rate the large regions as a whole and individual deposits within these regions. These matrices, and a more detailed explanation of the screening criteria (Table 1), deposits selected, and evaluations, are presented in a draft report now under technical review.[5] The Lake Superior region, as an example, is a region with potential for future study, based on the criteria utilized. It is extremely stable, is relatively free of faults, is far removed from volcanic centers, has low hydraulic gradients, has few mineral resources in crystalline hosts, and is covered by thick deposits of glacial sediments. Individual areas within each region were also evaluated, and many appear to have equally high potential, even through the region, as a whole, may have a lower rating. These areas are shown in Figure 2.

It is recommended that further studies be conducted in some of the selected regions. Much additional information should be collected and in-depth studies initiated to assess the suitability of individual deposits as possible repository sites.

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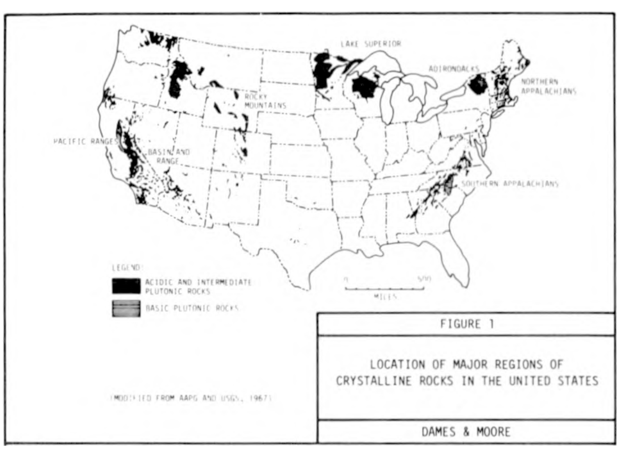

FIGURE 1. LOCATION OF MAJOR REGIONS OF CRYSTALLINE ROCKS IN THE UNITED STATES

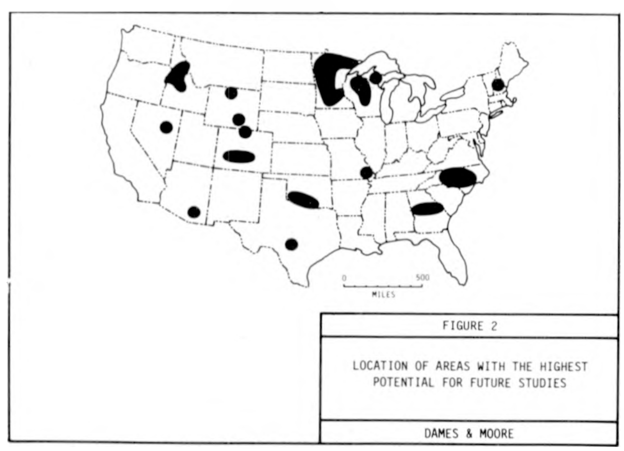

FIGURE 2. LOCATION OF AREAS WITH THE HIGHEST POTENTIAL FOR FUTURE STUDIES

TABLE 1. GEOLOGIC CRITERIA FOR A HIGH-LEVEL RADIOACTIVE WASTE REPOSITORY

Site Geometry	Adequate depth, thickness, and lateral extent of host media
Host-Rock Properties	Low permeability, porosity, and water content; adequate chemical, radiological, thermal, and physical and mechanical characteristics
Hydrology	Low groundwater gradients, adequate distance from utilization or discharge points
Tectonic Stability	Rates and amounts of uplift or subsidence cannot pose a threat to the physical integrity of the repository
Faulting	Located away from active faults which would adversely affect operational safety or geological containment
Volcanism	Cannot be sited in or near an area in which igneous or volcanic activity has occurred during the Quaternary period (present to 1-1/2 to 2 million years ago)
Seismicity	Must be located in regions expected to be remote from recorded or historic earthquakes of greater than "moderate" intensity
Relationship to Natural Resources	Located in area which will prevent preemptive utilization of potentially valuable resources

PRELIMINARY ASSESSMENT OF ARGILLACEOUS BASINS IN THE UNITED STATES

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Earth Resource Associates

Argillaceous rocks are presently viewed as a viable rock medium alternative to salt for the repository disposal of high-level radioactive wastes. Favorable characteristics include low permeability, relatively high plasticity, large ion-exchange capacity, and satisfactory response to the development of excavated-cavern space. Thick sequences of argillaceous rocks are found in many geologic settings[1], and most parts of the United States contain extensive deposits (Figure 1).

The objective of the current study, which involves a nationwide survey of the literature and other publicly available data sources, is to describe and characterize each of the major shale, mudstone, claystone, and certain argillite units in the conterminous states, and to assess the general suitability of each unit relative to the containment of radioactive waste. Recommendations about more detailed studies, including those of a site-specific nature, are to be developed for selected candidate units.

Assessment is being carried out for each laterally persistent argillaceous rock unit that is at least 75 m thick and in the depth range of 300 to 1000 m below the surface. Other important characteristics being evaluated for each formation include: physical properties; mineralogy; geochemistry, including the organic-matter fraction; geologic structures; and seismic and tectonic histories. Also being considered are the development of adjacent mineral resources, regional groundwater hydrology, and the extent of drilling, cavern construction, and mining into each rock unit.

Principal argillaceous rock units in the eastern United States include the late Ordovician shales of Ohio, Pennsylvania, and New York, and the Devonian-Mississippian shales of the Appalachian and Michigan basins.[2] These widespread units are locally at least 1000 m thick and involve clay mineralogies dominated by illite and chlorite. Other units undergoing examination are the Mississippian and Pennsylvanian shales in the Illinois basin and certain shales in some Triassic fault basins along the eastern seaboard.

The most significant argillaceous unit throughout the Gulf Coast region is the Paleocene-age Porters Creek clay, which is 150 to 350 m thick and contains montmorillonitic clay. Several other formations of Cretaceous and Tertiary age in this region may locally prove suitable for additional consideration as host rocks.

Shales of interest within the Midcontinent region are mainly Mississippian and Pennsylvanian units that extend from Iowa to West Texas. These shale units are typically less than 100 m thick, but in several basins of the region they attain thicknesses greater than 200 m. Mineral assemblages rich in illite and chlorite are typical of these shales.

Thick shales of Cretaceous age underlie much of the western Great Plains and Rocky Mountain regions. Shale units such as the Pierre and Mancos are typically 300 to 1500 m thick and consist mainly of montmorillonite and quartz. With the possible exception of the potential release of water from the montmorillonitic clay minerals upon heating (from the nuclear wastes), these units and their stratigraphic equivalents seem well qualified to be investigated in more detail.

West of the Rocky Mountains are several Devonian and Mississippian shales in the Basin and Range Provinces of Nevada, Utah, and Idaho. These shales are typically interbedded with limestone and sandstone, but are as much as 200 to 1000 m thick locally. Illite-chlorite clay suites predominate here.

Along the Pacific Coast are a series of Tertiary and Mesozoic shales and mudstones that are locally 500 m to more than 2000 m in thickness, but typically montmorillonitic. Because much of the region is subjected to periodic seismic activity, the potential of these units may be severely limited.

Other studies, such as the characterization of shale properties, thermo-mechanical testing of the Eleana formation and Conasauga shale, and gas-resource assessment of eastern Devonian shales, interrelate to the current investigation. Available data from these studies will be incorporated where appropriate.

During FY 1980, we will complete our regional assessment and characterization of each potential argillaceous rock unit in the United States, and submit a report that contains our findings.

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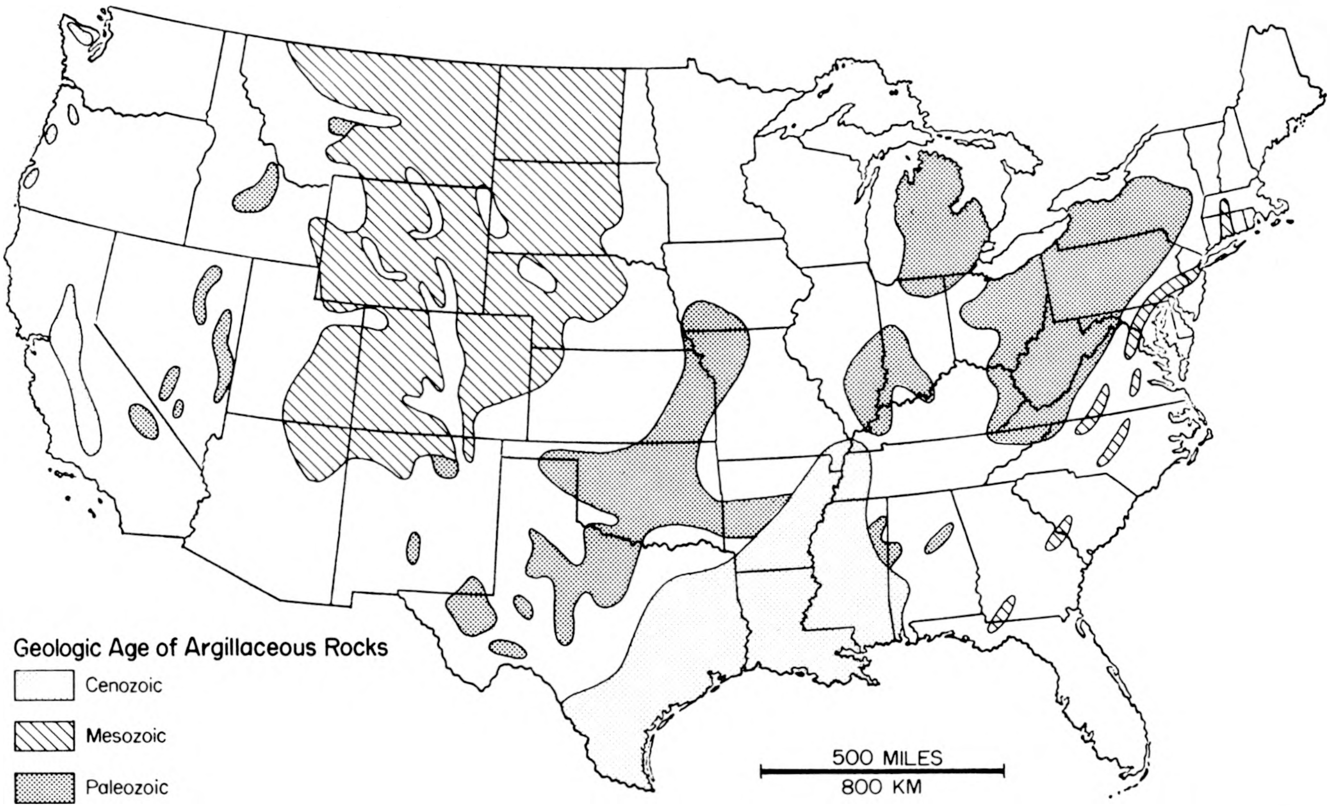

FIGURE 1. MAP SHOWING GENERALIZED OUTLINE OF AREAS UNDERLAIN BY THICK ARGILLACEOUS ROCK UNITS IN THE UNITED STATES Modified from Merewether and others.(1)

EVALUATION OF ARGILLACEOUS ROCK FOR NUCLEAR-WASTE CONTAINMENT

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Oak Ridge National Laboratory

Three broad classes of argillaceous rocks are being evaluated for nuclear-waste containment. These classes are defined as follows: (1) argillaceous rocks with no carbonaceous material and with little or no hydrous expanded clay mineral; (2) argillaceous rocks with carbonaceous material: and (3) argillaceous rocks with smectite as a major clay mineral constituent. Each class will be sampled over a range of values for the content of the major mineralogical constituents; i.e., clay, quartz, and calcite.

Samples of the argillaceous rocks will be obtained from sites where "typical" examples of these rocks can be found in well-characterized geological formations. The samples will consist of ten 33-m long (100-ft) cores of unweathered rock from a single site. Each sample will be divided into subsamples which will be characterized by: (1) complete chemical and mineralogical analyses, including measurement of the total volatile content of each sample over a range of temperatures from ambient to 900 C in 100 C steps; (2) response or reaction to large doses of gamma radiation; (3) measurement of the thermal properties of each sample at ambient temperature and 300 C; (4) measurement of the adsorptive properties of the rocks; (5) measurement of the mechanical properties at ambient temperature and 300 C; and (6) determination of the products of the interaction of the samples with simulated nuclear waste.

This project began in July, 1979, and will be funded in FY 1980 and FY 1981.

SESSION IIC
TECHNICAL STUDIES IN THE NWTS/ONWI PROCESS/EQUIPMENT DEVELOPMENT PROGRAM

IN SITU TESTING
OVERVIEW OF NWTS IN SITU TEST PROGRAM

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Office of Nuclear Waste Isolation

The accumulation of a sufficient technological data base to support the safe isolation of nuclear wastes in a geologic repository requires that many complex experiments be conducted in the laboratory, in pilot-plant facilities, and in situ in various candidate geologic formations. The development of in situ test data is necessary to provide verification of safety-performance-assessment models that are used in licensing and to compare with a much broader base of information that is obtained in the laboratory. To understand how in situ experimentation fits into the overall mission of development of a geologic nuclear waste repository, one must consider answers to the following questions:

(1) What is in situ testing?

(2) How does NWTS/ONWI determine what in situ information is needed?

(3) Who does the work?

(4) Where are in situ tests conducted?

(5) Who uses the information?

(6) What should be done in the future?

What Is In Situ Testing?

In situ testing for assessment of geologic nuclear waste repositories, is, by definition, conducting tests in the natural "in-place" environment of the particular geology being studied. In testing within a specific medium (e.g., rock salt, granite, or shale) one usually characterizes the various formations of interest by obtaining core samples of the formation and subjecting these to numerous laboratory tests to obtain data on chemical, physical, and mechanical properties, stratigraphy, hydrology, geochemistry, and waste/rock interactions. Then tests are conducted in selected formations, that are considered worthy of further study, with electrically heated canisters, chemical simulants, and actual wastes placed in the formation at expected waste/repository conditions. These types of tests are conducted near the earth's surface or below ground in mined tunnels. The reason for conducting some experiments in situ rather than relying solely upon laboratory data is that some parameters, such as local hydrology, overburden stress, and impact of mining, cannot be duplicated in the laboratory. Thus, final verification of important laboratory data and modeling predictions should be obtained in situ.

How Does NWTS/ONWI Determine What In Situ Information Is Needed?

The needs for in situ testing are determined from information provided by senior scientists, various government reports, congressional hearings, and technical meetings and workshops involving nuclear waste management. Representative examples of sources used to develop plans for in situ testing in different geologic formations are:

- Report to the President by the Interagency Review Group on Nuclear Waste Management
- National Academy of Sciences Committees
- DOE/DOI Earth Sciences Technical Plan
- NWTS in situ test needs workshops
- Professional society meetings.

Who Does the Work?

The subcontractor selection process is an important part of ONWI's role of project management for in situ testing and other parts of geologic repository development. Technically qualified people are sought from many universities, government laboratories, and various commercial firms. A competitive-bid contracting process is used whenever possible to find the best qualified organization at a competitive price.

Where Are Tests Conducted?

As indicated previously, in situ tests are conducted in candidate geologic formations determined from site exploration and laboratory studies. Usually, an abandoned mine or abandoned section of an operating mine is selected as a site for in situ tests if a representative repository environment can be found in or near the mine. Sometimes it is sufficient to conduct early exploratory tests in formations near the surface of the earth. Figure 1 shows in situ testing sites that are currently being used to evaluate salt, granites, shales, basalt, and tuff geologic formations.

Who Uses The Information?

The information and data that are obtained in various in situ tests are used primarily by scientists and engineers who are responsible for designing, performing safety assessments, and licensing geologic repositories for isolation of nuclear wastes. The data are used to develop and verify computer models used in design and safety analyses and to validate chemical, physical, and mechanical properties of the geology that were obtained in the laboratory. Also, the key results are made available to the public via the DOE information program.

What Should Be Done in the Future?

A required element in addressing technical issues by conducting in situ tests is the availability of appropriate test facilities. Some test facilities should be located where actual high-level waste forms can be exposed to representative repository geologic environments. Thus, it is planned to develop in situ test facilities in salt, granite, basalt, and other candidate geologic formations if appropriate. The specific type of information needed will depend on the particular isolation properties of each geology. However, a general set of in situ test needs has been identified as follows:

- Nuclide leaching and migration
- Canister corrosion
- Evaluation of engineered barriers
- Water flow in and around the repository
- Gas generation
- Mine stability/design criteria
- Evaluation of repository sealing.

Planning of specific in situ tests is currently under way to allow verification of computer models that provide a performance safety assessment of these factors in the isolation of nuclear wastes in geologic repositories. The papers presented in the remainder of this session describe the in situ information that is currently being obtained by NWTS/ONWI in salt, granite, and shale.

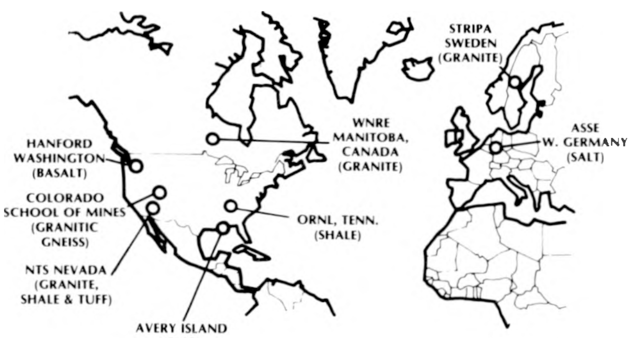

FIGURE 1. WHERE ARE IN SITU TESTS CONDUCTED?

DOME-SALT THERMOMECHANICAL EXPERIMENTS AT AVERY ISLAND, LOUISIANA

L. L. Van Sambeek
RE/SPEC, Inc.

Three in situ electrical heater tests have been in operation within the Avery Island salt mine since June, 1978. The overall purpose of the heater tests is to provide quantitative data concerning the response of domal salt to the presence of a heat source. The data will be used in the determination of whether or not domal salt behaves similarly to bedded salt; i.e., is the Project Salt Vault experience applicable to domal salt? Three heater test installations were made in the floor of the mine and data are being gathered on temperatures, heat fluxes, displacements, stresses, and mine environment conditions.

Preliminary conditions concerning the heater tests do not show major deviations in the thermomechanical response of the Avery Island domal salt when compared with that observed in Project Salt Vault. Temperature measurements, however, have yielded data which indicate the in situ thermal conductivity is probably as much as 10 percent higher than that measured on core samples in the laboratory. The heater tests are continuing with probable termination during the time period of October to February in FY 1980. Additional in situ tests will be initiated at Avery Island to explore salt response at expected repository conditions.

Description

The primary activities of the in situ testing project at the Avery Island mine operated by the International Salt Company, part of Akzona, Incorporated, in southwestern Louisiana involve three heater tests. The three heater tests employ simulated, electrically heated, waste canisters placed in sleeved boreholes in the floor of the mine. Installation of the test equipment was begun in January, 1978; all three tests were in operation by June, 1978. The overall physical makeup of each test is similar; the significant differences concern the simulated waste-canister power level and the presence of crushed backfill salt around the sleeve. A brief description of each of the three tests follows:

Site A: This experiment involves a single, central heater performing at a 6-kw power level, situated in a carbon steel sleeve and residing between depths of 3.0 and 5.4 meters below the salt floor level. Located around the heater installation are 150 temperature sensors, 50 floor-heave stations, 7 multiple-point extensometers, and 3 heat-flux meters. The test is essentially an overdrive test due to its high power level; the large number of data-gathering locations is for the purpose of providing an axisymmetric test case for numerical modeling studies.

Site B: This test involves an installation similar to that of Site A, but operating at 3 kw, one-half the power level of Site A. Only a moderate amount of instrumentation is included; specifically, 65 temperature sensors, 50 floor-heave stations, 3 multipoint extensometers, and provisions for manually monitoring borehole closure. Site B is also applicable as an axisymmetric test case for numerical modeling; however, obtaining measurements of borehole closure and corrosion by-product gas generation at a realistic power level is its primary purpose.

Site C: This is a multiple-heater installation where a single central heater, operated at a constant power level of 4 kw, is surrounded by eight smaller peripheral heaters, operated at an increasing power level, to produce a unit-cell configuration. The annulus between the sleeve and borehole in the salt floor is also backfilled with crushed salt. Instrumentation at Site C is similar to that employed in the Site B test, with the addition of thermocouples located in the backfill. The primary objective of the test is to provide data concerning sleeve design for backfill situations. The unit-cell configuration simulates heat input from adjacent canisters and will provide for increased energy in the salt, enhancing corrosion potential and thermomechanical response.

Preliminary Results

The measured temperatures from each of the tests can be divided into two groups: the heater-assembly temperatures and temperatures measured in the salt. Figure 1 presents the measured temperatures of the heater-assembly components after 300 days of heating, together with an extrapolated salt borehole-wall temperature as appropriate. The temperatures for Sites A and B are essentially steady-state, while those for Site C continue to increase with increasing power to the peripheral heaters. Figure 2 presents temperature profiles at various elevations from the Site A heater test after 300 days of heating. A complete presentation of temperature data can be found in Reference (1). The measured temperatures are in reasonable agreement with those calculated by Wagner and Van Sambeek[2] when using a thermal conductivity 20 percent greater than measured on core samples in the laboratory. Displacements are being measured by both multiple-point extensometers and optical-level survey. Figure 3 presents a representation of the floor uplift and differential expansion of the salt around the Site A heater test at a radial distance of 3 m from the heater centerline. The measured displacements are all within the range that was calculated during the design of the experiment.

The measurement of induced stress using vibration-wire stress meters has not been totally satisfactory. Qualitatively, the data indicate that the thermally induced stress was sensed by the gages; however, as temperatures increased near the gages, the response and its interpretation became inconsistent.

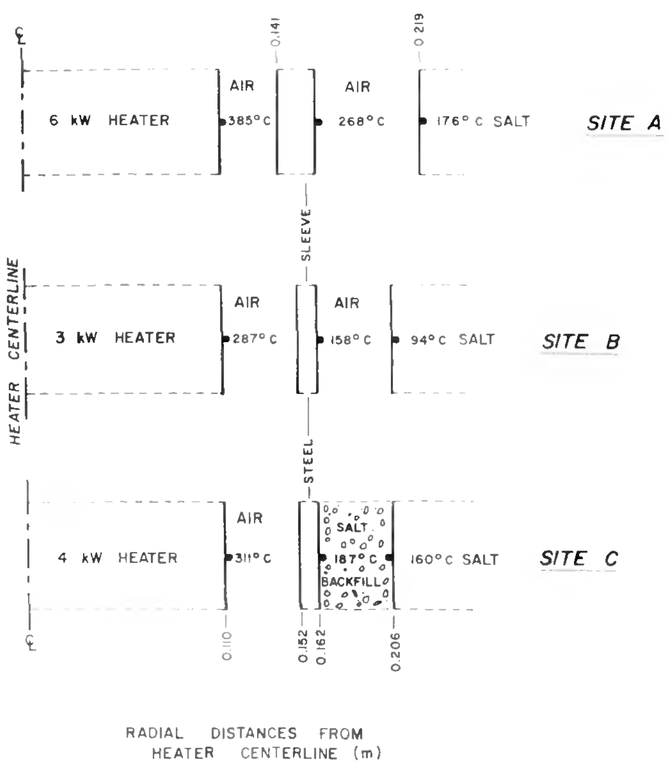

FIGURE 1. MEASURED TEMPERATURES OF HEATER ASSEMBLY COMPONENTS AFTER 300 DAYS OF HEATING

Future Activities

The three heater tests will be completed early in FY 1980. The Site A test will be monitored for cool-down temperature data, and, subsequently, salt core samples will be taken for post-test examination. Site B will be retrofitted with a moisture-collection system prior to being turned off, to examine whether moisture enters the borehole during cool down. The test at Site C will be completed during FY 1980.

A series of bench-scale tests, where repository conditions of stress and temperature will be produced, will be performed at the test site during FY 1980. The tests will address such considerations as borehole closure, backfill consolidation, and fracture healing. An additional set of measurements will be performed to quantify the change in permeability of salt due to heating.

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(2) Wagner, R. A. and Van Sambeek, L. L., "Finite Element Analysis and Prediction of the Response of the Site A Heater Test at Avery Island" (draft under review).

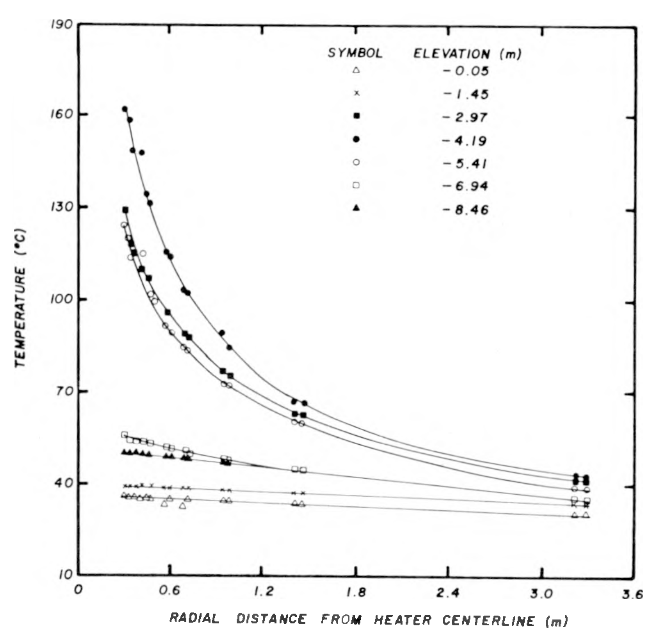

FIGURE 2. TEMPERATURE PROFILES ABOUT THE HEATER TEST "A" AFTER 300 DAYS OF HEATING AT 6-kW POWER LEVEL

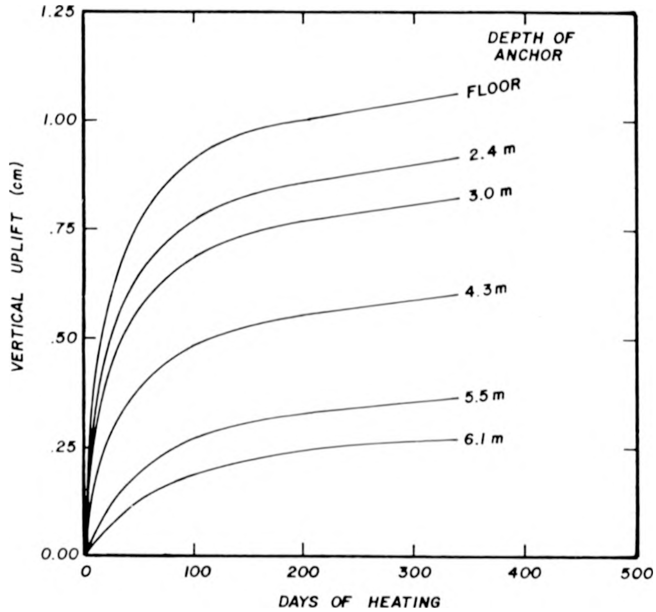

FIGURE 3. VERTICAL UPLIFT MEASURED BY AN EXTENSOMETER AT A RADIAL DISTANCE OF 3 m FROM HEATER CENTERLINE

DOME-SALT BRINE-MIGRATION EXPERIMENTS AT AVERY ISLAND, LOUISIANA

W. B. Krause and P. F. Gnirk
RE/SPEC Inc.

A series of in situ brine-movement experiments is under way in dome salt on the 168 m (550 ft) level of the Avery Island mine (International Salt Company, a part of Akzona, Inc.) in southwestern Louisiana. The objectives of the experiments are to examine the movement of synthetic brine in salt in a temperature field induced by emplaced electrical heaters, and to develop requisite and more precise measurement techniques and procedures for use in future natural-brine-migration experiments. In a more general sense, these experiments address, in a preliminary fashion, the question of the extent of brine movements in the vicinity of emplaced radioactive waste canisters as induced by thermal perturbations in the salt. The resolution of this question has considerable impact on the design and emplacement of backfill materials and other protective measures for isolation of waste canisters in a repository from natural brine in geologic salt.

Details of the Experiments

As shown in Figures 1 and 2, the basic experimental configuration consists of a borehole with a 1-kw electrical heater which is centrally located within a pattern of other boreholes containing thermocouples and synthetic brine. By proper selection of the heater borehole diameter, the heat flux on the borehole wall from a 1-kw heater will be the same as that for a 3-kw waste canister in a 40.6-cm-diameter borehole. The synthetic brine is "mixed" with glass beads in order to minimize thermal convection within the borehole. The brine is tagged with deuterium, so that the extent of brine movement within the salt after a specified heating period can be determined from drilled core samples.

Three separate tests will be performed for the purpose of evaluating:

(1) The areal extent of brine movement for the induced thermal gradient

(2) The rate of brine inflow into the heater hole for the induced thermal gradient

(3) The relationship between in situ salt permeability and Items (1) and (2)

(4) The influence of thermally induced microfracturing in salt around the heater emplacement on Items (1) and (2).

The three distinct experiments may be described as follows[1]:

(1) **TEST AB: Natural Brine Movement Under Ambient-Temperature Conditions.** This control experiment is designed to evaluate the movement of any natural brine in the dome salt as a consequence of stress perturbations induced by drilling the borehole configuration. The salt will not be subjected to an induced thermal field with an electrical heater.

(2) **TEST NB: Natural Brine Movement Under Elevated-Temperature Conditions**. This control experiment is designed to evaluate the movement of any natural brine inclusions in the dome salt under elevated-temperature conditions. Use will be made of a 1-kw electrical heater, with thermocouples placed in boreholes in the salt. Periodic sampling of vapors and liquids in the heater hole will be performed to determine when and if an influx of natural brine occurs. Core specimens will be obtained in the vicinity of the boreholes and by overcoring the heater hole after completion of the experiment for analysis of the mechanical condition of the salt and of brine concentrations.

(3) **TEST SB: Synthetic Brine Movement Under Elevated-Temperature Conditions.** By use of a 1-kw electrical heater, the temperature in the surrounding salt will be raised to 75 percent of steady-state. At this thermal condition, synthetic brine and glass beads will be introduced into the "brine" boreholes and pressurized (345 to 690 kPa). Temperatures in the salt and influx of brine to the heater hole will be monitored as heating is continued. The synthetic brine level in the boreholes will be maintained constant by removal or replenishment of the brine as required. As described in Test NB, salt specimens for analysis will be obtained by coring after termination of the experiment.

These three experiments were preceded by a "prototype" test (same geometry and physical dimenions) to determine:

(1) Accuracy of drilling boreholes to designated tolerances

(2) In situ permeability of salt

(3) Extent of microfracturing in salt as a consequence of drilling (and stress concentrations due to borehole layout geometry) and heating

(4) Effectiveness of borehole seals in the brine holes.

Heat Transfer Analysis

By use of finite-element models, the temperature field in the salt around the heated borehole was determined[2]. The appropriate geometry is shown in Figure 3 and the calculated temperature and temperature-gradient profiles are given in Figures 4 and 5, respectively.

Future Work

The prototype test was initiated in early August, 1979, and the three experiments are in progress. Evaluation of the test data is scheduled to be completed during the first quarter of CY 1980.

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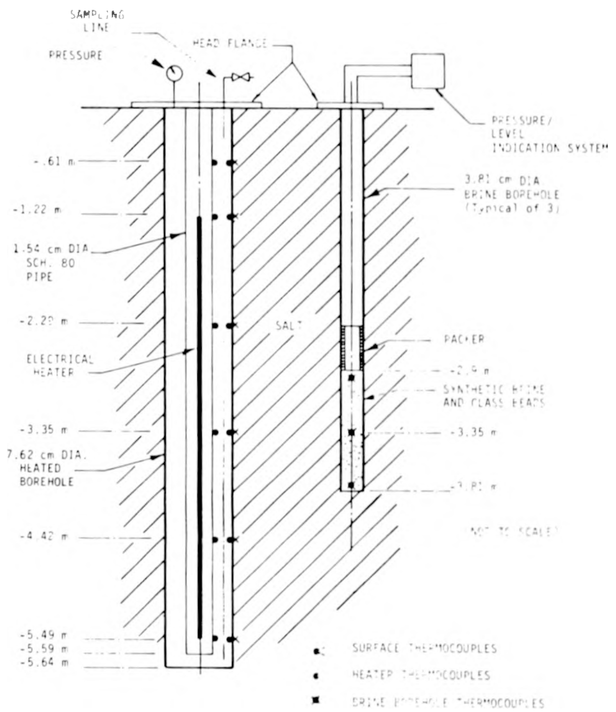

FIGURE 1. CONFIGURATION SHOWING HEATED BOREHOLE AND BRINE BOREHOLE

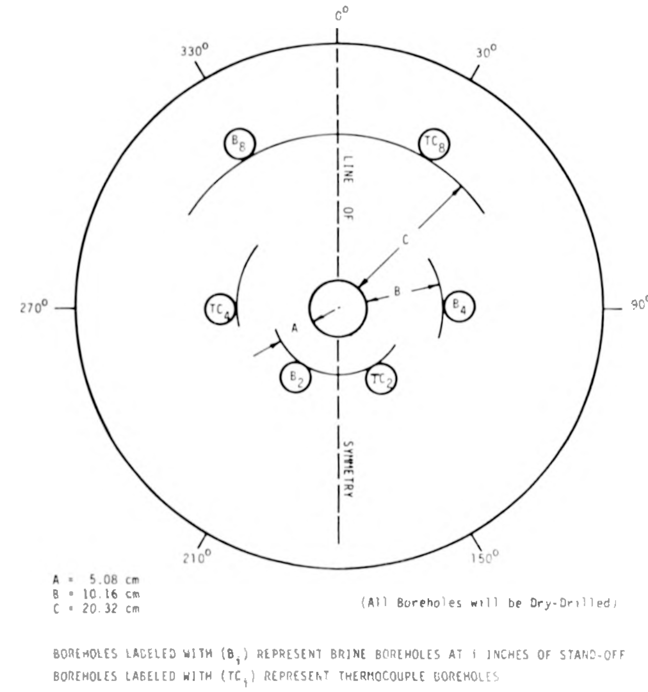

FIGURE 2. LAYOUT OF BOREHOLES FOR BRINE MOVEMENT EXPERIMENT AT AVERY ISLAND

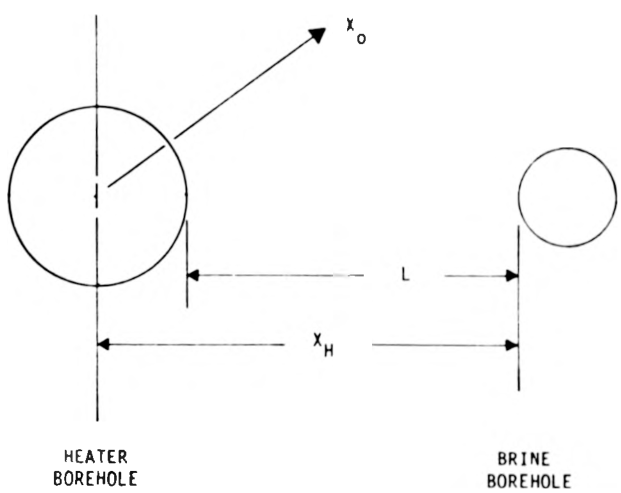

FIGURE 3. DEFINITION OF DISTANCES USED IN HEAT TRANSFER ANALYSIS

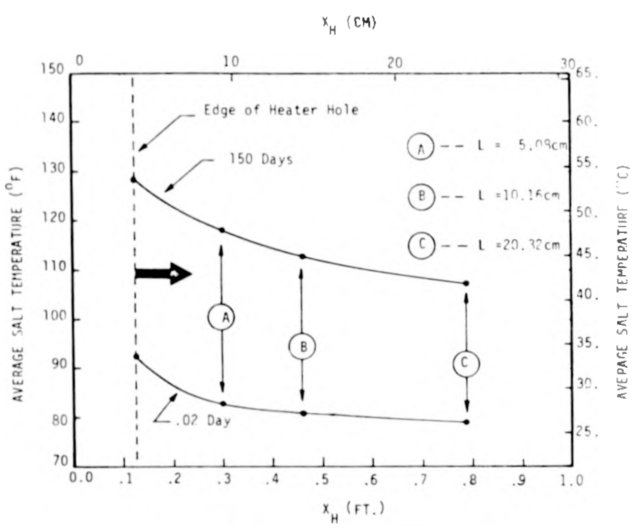

FIGURE 4. RANGE OF TEMPERATURES AS A FUNCTION OF DISTANCE FROM CENTERLINE OF HEATER TO EDGE OF BRINE BOREHOLES

Midplane elevation = -3.35 m. Heat input level is 1 kW.
K = 5.94 W/m-°C, C_p = .85kJ/kg-°C.

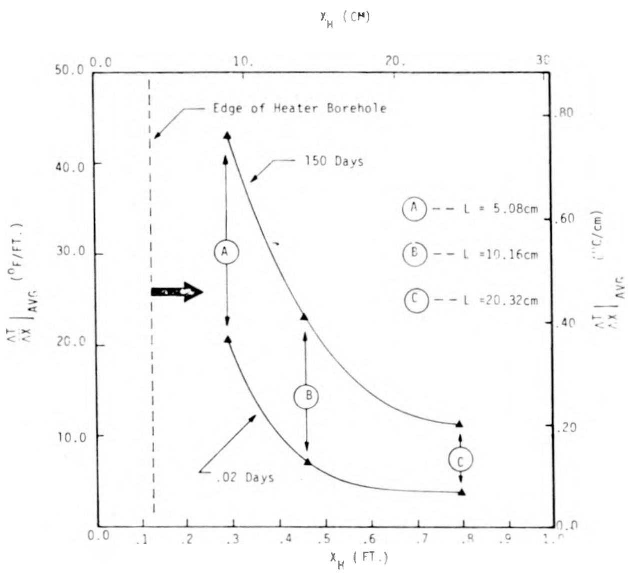

FIGURE 5. RANGE OF GRADIENTS AS A FUNCTION OF DISTANCE FROM CENTERLINE OF HEATER TO EDGE OF BRINE BOREHOLES

Midplane elevation = -3.35 m. Heat input level is 1 KW.
$K = 5.94\ W/m\text{-}^{o}C$, $C_p = .85\ kJ/kg\text{-}^{o}C$.

THERMOMECHANICAL EXPERIMENTS IN GRANITE AT STRIPA, SWEDEN

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Geologic disposal of nuclear wastes within excavations made at depths in suitable media has long been [1], and continues to be [2], favored as currently the most practicable means of isolating them from the biosphere over the long term. Although a significant body of experience exists concerning underground excavation, it does not include the effects of heat generation on the excavations and the geologic media. Experiments to assess some of these effects have been done for salt in Project Salt Vault [3] but it is now agreed that other media should be examined [2]. Access to tunnels driven into the granite country rock 340 meters below the surface, adjacent to a defunct iron ore mine at Stripa, Sweden, provided a unique opportunity for doing experiments in granite without delay and at minimal cost. The test site was developed at a depth where conditions of stress, jointing, groundwater, and other factors associated with depth are similar to those likely to be encountered at the site of an actual waste repository.

The disposal of high-level nuclear waste deep underground will result in the geologic media in the vicinity of such a repository undergoing a thermal pulse. This pulse will induce thermomechanical displacements and stresses in the rock. In general, these displacements will be directed away from the source of the heat while the temperature is increasing, then tending to return as the temperature decreases. Likewise, the thermomechanical stresses will result in the addition to the virgin state of stress in the rock of compressive stresses within the heated zone and the addition of deviatorial stresses, including tension, outside of it.

Transport by groundwater is the most probable mechanism by which components of the wastes may find their way back to the biosphere. The intrinsic permeability of many granites is so low that the only hydraulic conduits of concern arise from joints and fractures in masses of such rock. Clearly, the thermomechanical perturbations may have significant effects on the hydraulic transmissivity of such features. Accordingly, it is necessary that the effects of these perturbations be understood if the utility of a geologic formation as a site for a potential waste repository is to be evaluated properly. Furthermore, the design of a repository and predictions concerning its performance in the long term cannot be done without such an understanding.[4]

The value of field experiments depends upon their success in obtaining a degree of understanding of the phenomena involved that is sufficient to enable the results to be transferred to other sites where actual repositories may be built. Accordingly, the program involves the collection of sufficient field and laboratory data to ensure that their analysis will provide either a high degree of understanding of the behavior of the rock mass or an identification and definition of those crucial issues requiring further research.

Three different thermomechanical experiments are under way at Stripa.[5] The first is designed to study the short-term near-field effects around an electrical heater simulating a full-size canister of reprocessed high-level waste. The second is a similar experiment designed to study the long-term near-field effects. The third is a time-scaled experiment designed to simulate the interaction between adjacent canisters over a period equivalent to about two decades, using the quadratic relationship between time and distance in linear thermoelasticity.[6]

Based on the theory of linear thermoelasticity and properties of the granite measured in conventional small-scale laboratory tests, the results of all three experiments — namely, the expected temperature, displacement, and stress fields as functions of time —have been predicted in advance of the collection of field data. The in situ data are being collected in such a way as to allow comparisons between theoretical predictions and underground measurements to be made continuously during the experiment. To date, the comparisons have shown that the use of simple linear heat conduction provides an adequate prediction of the temperature fields around the three experiments (Figures 1 and 2). According to the theory of linear thermoelasticity, displacements should be related to temperature fields by a simple factor, $\alpha[(1+\nu)/(1-\nu)]$, where α = the linear coefficient of thermal expansion of the rock, and ν = Poisson's ratio for the rock. All the measured displacements differ significantly from predicted values in two different ways (Figure 3). First, initial displacements, of the order 100 μm per meter, are highly nonlinear, reflecting possibly the effects of fracture joints in the rock. Second, greater displacements than these appear to be linear but to have a magnitude only about half that expected from values derived from simple laboratory measurements. Likewise, the stresses appear to have values different from that given by the temperature field and a factor $\alpha E/(1-\nu)$, where E = the Youngs' modulus of the rock and the other symbols are as defined above (Figure 4).[7-8]

The disparities between measurement and predictions using simple theory and laboratory data should not be regarded as evidence of a lack of predictive capability but, rather, as a means for identifying and understanding the important differences in behavior between a rock mass and laboratory specimens of rock.

The power levels of the two full-scale heaters are 3.6 kw and 4 kw. These heat loads simulate the expected decay heat for reprocessed fuel 5 years and 3.5 years after discharge from the reactor.[9] These power levels, together with the peripheral heaters used in the second stage of the 5-kw heater experiment to simulate the effects of increasing the temperature of the rock containing a waste canister, have produced thermal stresses on the walls of the boreholes containing these heaters below, at, and above those sufficient to cause decrepitation of the granite.[10] This has enabled the conditions causing decrepitation to be defined (Figure 5).

The core from every instrumentation hole at Stripa was logged during drilling and stored for future reference or use. A laboratory test program is under way to obtain the thermomechanical properties of speciments of the core taken from the same hole in which the measurements of displacements and stress have been made. These tests will incorporate a range of hydrostatic and deviatorial stresses and temperatures covering and exceeding those to which this rock has been subjected in the field. They will include specimens of intact and jointed rock. These laboratory values will then be used to refine the predictive models, incorporating such nonlinear properties as may be revealed.

Finally, starting in June, 1979, power to the heaters has been turned off. Continuous measurements are being made during the cool-down period, as they were during the heat-up period. It is expected that substantial additional information, especially concerning nonlinear phenomena, will be obtained by observing hysteresis over a full thermal cycle.

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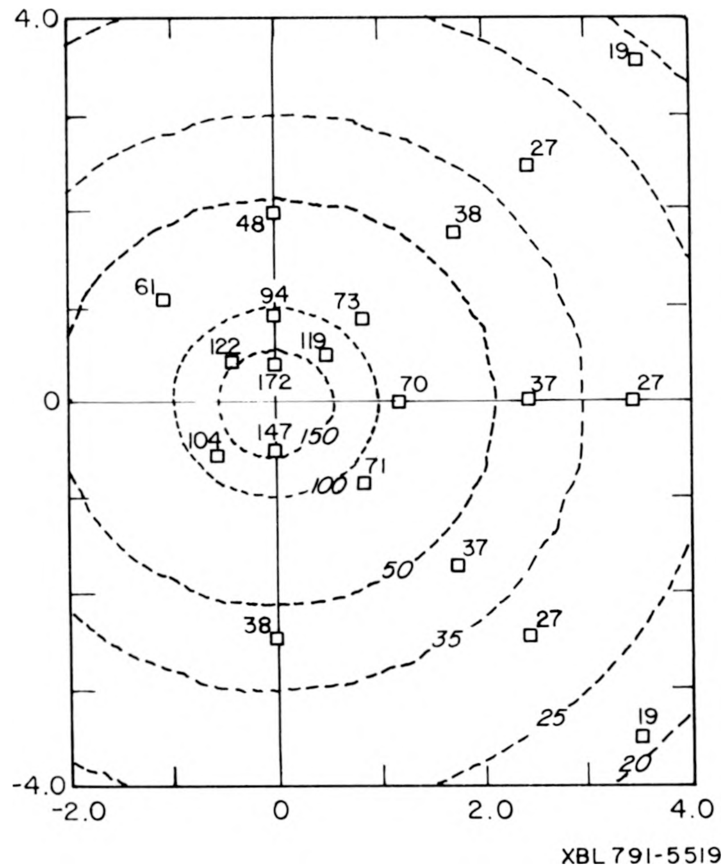

FIGURE 1. A PLAN OF THE HORIZONTAL PLANE THROUGH THE MIDDLE OF THE 5-kW FULL-SCALE HEATER SHOWING THE PREDICTED ISOTHERMS (DASHED LINES) AND ACTUAL TEMPERATURES (SQUARES WITH NUMBERS) MEASURED ALONG DIFFERENT DIRECTION AT 190 DAYS AFTER THE START OF HEATING

Note the relatively good correlation between measurement and prediction and the high degree of radial symmetry (scales for the X- and Y-axes are shown in meters).

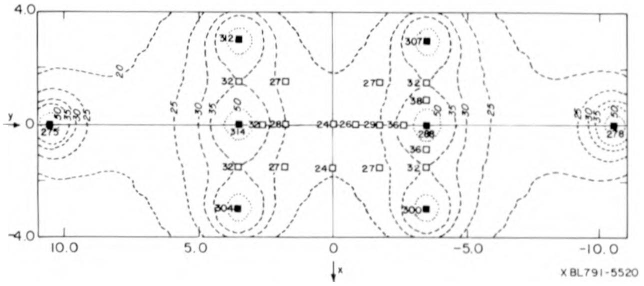

FIGURE 2. A PLAN OF THE HORIZONTAL PLANE THROUGH THE MIDDLE OF THE ARRAY OF 8 TIME-SCALED HEATERS SHOWING THE PREDICTED ISOTHERMS (DASHED LINES) AND MEASURED TEMPERATURES (SQUARES WITH NUMBERS) AT 190 DAYS AFTER THE START OF THE EXPERIMENT

Actual distances between the heaters (black squares) are given in meters along the X- and Y-axes. The spacing between these heaters corresponds to 7 meters and 22 meters for full-scale heaters and the temperatures to those at 1938 days (5.3 years) because of the time scaling.

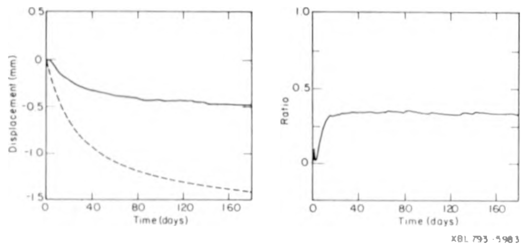

FIGURE 3. AN EXAMPLE SHOWING THE PREDICTED (DASHED LINES) AND MEASURED (SOLID LINES) DISPLACEMENTS BETWEEN ANCHOR POINTS SITUATED 3 METERS ABOVE AND BELOW THE MIDPLANE OF THE 5-kW HEATER AND AT A RADIAL DISTANCE OF 2 METERS AS A FUNCTION OF TIME (LEFT), TOGETHER WITH A PLOT OF THE RATIO BETWEEN THE MEASURED AND PREDICTED VALUES AS A FUNCTION OF TIME (RIGHT)

Note the initial nonlinearity of the displacements and their subsequent linear, but lower than predicted, behavior revealed by the right-hand plot.

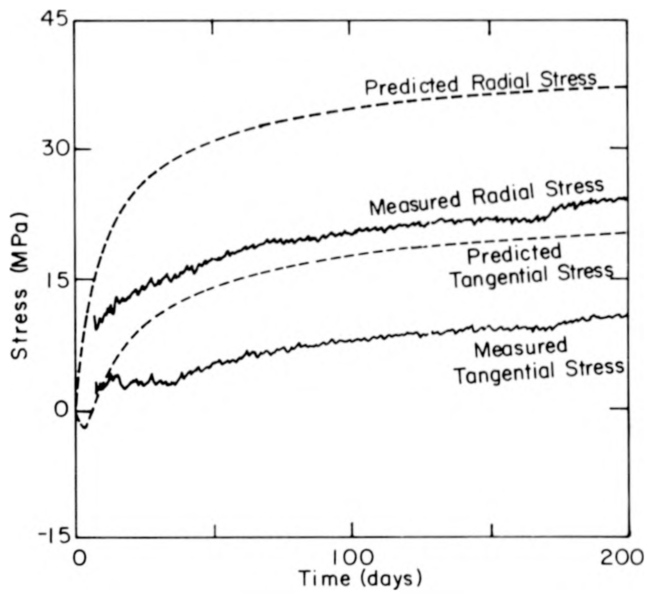

FIGURE 4. AN EXAMPLE OF THE PREDICTED CHANGES (DASHED LINES) AND MEASURED CHANGES (SOLID LINES) IN A STRESS AS A FUNCTION OF TIME, INFERRED FROM A VIBRATING WIRE BOREHOLE STRAIN GAGE LOCATED 0.85 METER ABOVE THE MIDPLANE AND 1.5 METERS RADIALLY FROM THE 5-kW FULL-SCALE HEATER

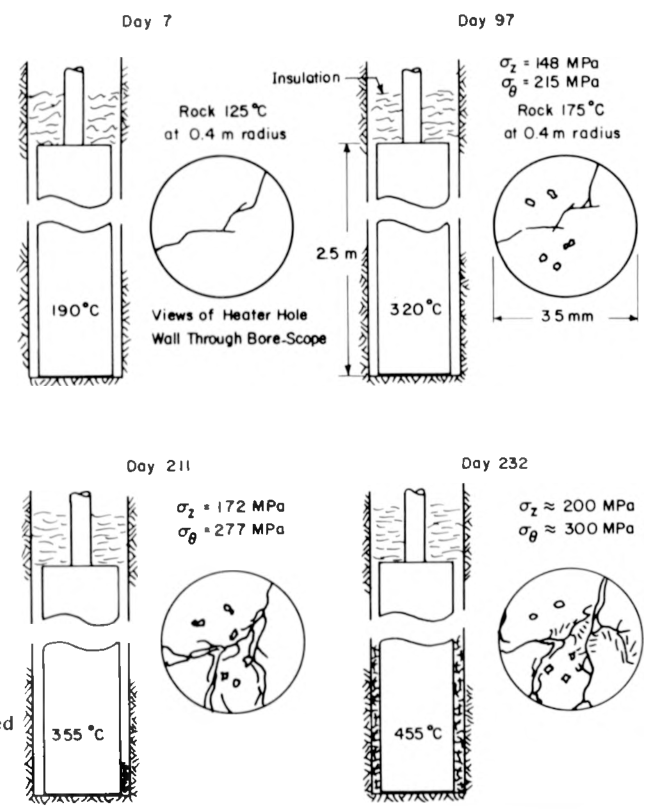

FIGURE 5. AXIAL SECTIONS THROUGH THE 5-kW FULL-SCALE HEATER WITH SKETCHES OF BORE-SCOPE VIEWS OF PORTIONS OF THE HOLE CONTAINING THIS HEATER, ILLUSTRATING THE DECREPITATION OF THE BOREHOLE WALL CAUSED BY THERMALLY INDUCED STRESSES, THE MAGNITUDES OF WHICH ARE GIVEN

Note that the gross decrepitation caused by the additional stress induced by turning on the 8 peripheral heaters on day 204 impeded the radiant heat transfer from the heater to the rock causing the temperature of the heater to increase by about 100° C.

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FRACTURE-HYDROLOGY AND GEOPHYSICAL STUDIES AT THE STRIPA MINE, SWEDEN

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Fracture-hydrology studies at a given site must be based on a detailed study of the fracture system in which one attempts to describe the three-dimensional fracture network—fracture spacing, orientation, continuity, and degree of interconnection. A description of the mechanical characteristics of the fracture enables one to assess the effect of in situ stresses on flow through the fracture system. In addition, a detailed knowledge of the fracture system is required if one is to be able to interpret and predict the effects of the thermal-mechanical loads produced by the waste decay on the stability and hydraulic characteristics of the rock mass in the immediate vicinity of a repository.

The primary objectives of the fracture-hydrology study at Stripa are (1) to assess the applicability of both new and existing tools and techniques in measuring the hydraulic characteristics of fractured granitic rocks and (2) to contribute to our understanding of the role of fractures in the movement of water through such rock masses. Hence, these studies support the NWTS/ONWI long-term goal of determining the potential for removal and transport of radioactive materials by the groundwater system to the biosphere from a repository located in fractured granitic rocks.

The LBL fracture-hydrology program at Stripa, Sweden, is summarized in Figure 1 and consists of five basic field activities: (1) assessing directional permeabilities, (2) geochemical and isotope studies, (3) tracer studies, (4) an underground ventilation experiment to measure rock-mass seepage, and (5) pump tests. All five components are integrated and supported where needed by mathematical modeling. The fracture-hydrology field program is supported by a borehole geophysical logging program. Both the hydrology and geophysical investigations are supported by laboratory programs.

Assessing Directional Permeabilities

The basic approach to assessing the directional permeabilities of the fractured granite at Stripa has been discussed by Gale and Witherspoon.[1] This approach, based on earlier work [2-4], requires a detailed knowledge of fractured orientations, spacing, and degree of fracture interconnection. These data have been obtained by mapping the fractures in the surface outcrops, in the walls of the subsurface excavations, and those encountered in the drill core. Figure 2 shows the distribution of surface outcrops of granite and leptite, a plan projection of the subsurface test excavations and the location of the surface and subsurface hydrology boreholes. The arrangement of boreholes in the north end of the tunnel, in the section being used for the ventilation experiment, is shown in Figure 3.

The surface boreholes consist of six water-table wells, WT-1 through WT-6; a pump-test well, WT-7; and three long inclined boreholes, SBH-1, SBH-2, and SBH-3 (Figure 2). The drill cores from SBH-1, SBH-2, and SBH-3 have been reconstructed and true fracture orientations calculated. Nearly all of the boreholes have been tested with a double packer assembly using a fixed and variable packer spacing. Typical data for the subsurface boreholes are given in Figure 4. The general geological log, distribution of fracture zones, RQD values, and in situ fluid pressures for SBH-1 are given in Figure 5. The low permeabilities encountered have required the development of equipment for measuring flow rates as low as 0.01 to 0.001 ml/min. Supporting laboratory studies include a study of the stress permeability characteristics of induced and natural fractures in 15-cm-diameter cores and an assessment of the effects of packer compliance on pressure-pulse tests. Pressure-pulse tests have been carried out in five of the boreholes in the ventilation drift. Both the laboratory studies and the field work show the overriding influence of temperature variations, small packer leaks, and packer stiffness on pressure-pulse test results.

Analysis of pump-test data[5] from WT-7, 100 m in depth, suggests hydraulic conductivity values of about 10^{-5} cm/sec.

Groundwater Geochemistry And Isotopes

Groundwater geochemical data provide not only basic data for use in chemical-transport problems, but also provide one of the few definitive checks on flow-velocity calculations resulting from hydrogeological studies. Age dating and paleoenvironmental analyses are the critical data that are needed.

We have examined the applicability of this approach through some very careful studies at Stripa.[6] Groundwater samples were collected from shallow and private wells, surface boreholes, and boreholes drilled from the 330-m and 410-m mine levels. These samples have been analyzed for their major ion chemistry, dissolved gases, and environmental isotope contents in order to describe their origin, age, and geochemical history.[6, 7]

Preliminary ^{14}C age dates indicate that waters presently discharging at the 410-m levels were recharged more than 20,000 years ago (Figure 6). Although the data indicate somewhat younger dates at depths below the 330-m level, this may be due to atmospheric contamination or the in situ generation of ^{14}C due to the relatively high radionuclide content in the granite. A second set of water samples has recently been taken to check these points. Helium contents of the waters and results from $^{234}U/^{238}U$ activity ratios cannot yet be interpreted quantitatively, but qualitatively the data indicate that significantly older waters exist at the 410-m level relative to those at the 330-m level. Confirmation will have to be obtained through ^{14}C analyses.

Oxygen-18 and chloride contents give clear evidence that modern surface waters are not actively recharging from fracture systems encountered at the 330-m and 410-m levels. This is shown in Figure 7, where there are distinct differences evident between these geochemical results for samples collected from the shallow systems (surface waters and shallow wells) and from the deep fracture systems (330-410 m). The ^{18}O data on Figure 7 can be interpreted as indicating that the deep groundwaters recharged under cooler climatic conditions than exist today. No tritium was found in actively discharging groundwaters at the 330-m and 410-m levels, which further confirms the present lack of communication from the surface to the depths of the deep fracture systems.

An important aspect of the chemistry of these waters is that the pH rises to values > 9.5 at the excavation levels and below. Carbonate contents decrease with depth but Cl^-, Ca^{++}, Na^+, and SO_4^- increase. These changes are determined by rock-water interactions and the possible admixtures of minor amounts of fossil sea water.

Ventilation Experiment

Results of an investigation of a number of underground mines suggested that the walls of "dry" mines may appear to be dry because the permeability of the rock is so low that the mine ventilation system removes all the inflow as water vapor.[8] Hence, by measuring the amount of water vapor being removed by the ventilation system in an isolated section of the mine, a measure of the total water inflow can be obtained. In addition, by knowing the pore pressure distribution in the vicinity of the test zone, the rock-mass permeability can be calculated.

At Stripa, a 30-m section of the end of the test drift has been sealed with a bulkhead, and a ventilation or macropermeability experiment[9] is under way. Piezometer systems have been installed in each of the 15 boreholes drilled from this section of the drift (Figure 3), and the resulting fluid pressures and flowrates, determined from changes in the humidity of incoming and outgoing air, are expected to provide a rock-mass permeability for direct comparison with the directional permeability values calculated from injection tests and fracture data. This should provide permeability results on one of the largest masses of fractured rock ($\sim 10^6$ m^3) that has ever been tested.

Geophysical Measurements

The geophysical program at Stripa [10] has thus far concentrated on borehole tests for the following: definition of fractures, locating permeable zones, determining chemical and physical inhomogeneities, and quantitative assessment of water content and porosity. Seven different measurements have been obtained: neutron, gamma-gamma, resistivity, gamma-ray, sonic, caliper, and temperature. Gamma-ray logs show that Stripa granite is chemically heterogeneous and contains high levels of uranium and thorium. The most effective technique for fracture definition has been the borehole sonic tool. Pronounced sonic waveform anomalies are associated with major fracture zones; however, only a small percentage of fractures logged in cores or by television produced anomalous waveform events. The suite of logs for a portion of SBH-1 is shown in Figure 8 which shows the correspondence of geophysical results with fractures logged from core samples. Eleven zones of relatively high permeability have been predicted in SBH-1 based on sonic data and temperature logs.

Planned Activities in FY 1980

Injection testing of the initial group of boreholes will be completed by December, 1979. A multipoint piezometer will be installed in SBH-1. Significant efforts will be devoted to data analysis, tracer tests, additional geochemistry-isotope studies, completion of the ventilation experiment, and numerical modeling efforts. Geophysical work will include logging of SBH-2 and SBH-3. Injection tests will be carried out in conjunction with hydraulic fracturing experiments.

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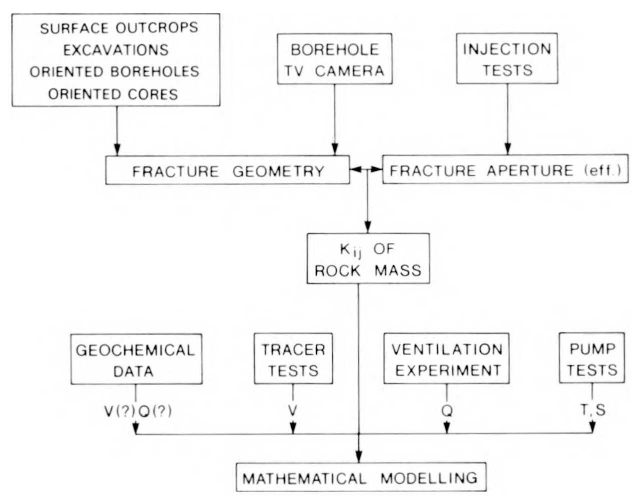

FIGURE 1. OVERALL APPROACH TO DETERMINATION OF FRACTURE PERMEABILITY[1]

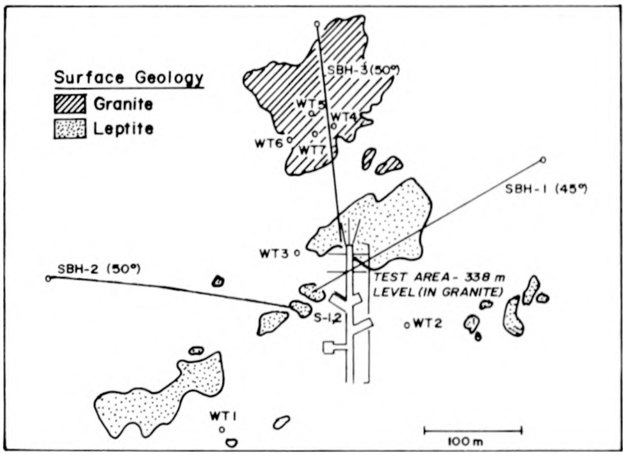

FIGURE 2. AREAL GEOLOGIC MAP SHOWING GENERAL LAYOUT OF SWEEDISH-AMERICAN COOPERATIVE PROJECT AT STRIPA, SWEDEN

Glacial deposits prevail except where leptite and granite outcrop.[1]

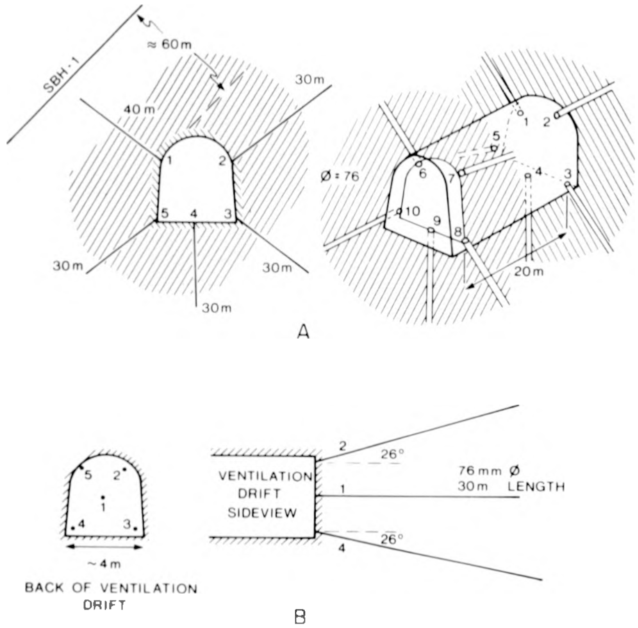

FIGURE 3. ARRANGEMENT OF SUBSURFACE BOREHOLES IN THE TUNNEL SECTION USED FOR VENTILATION EXPERIMENT[1]

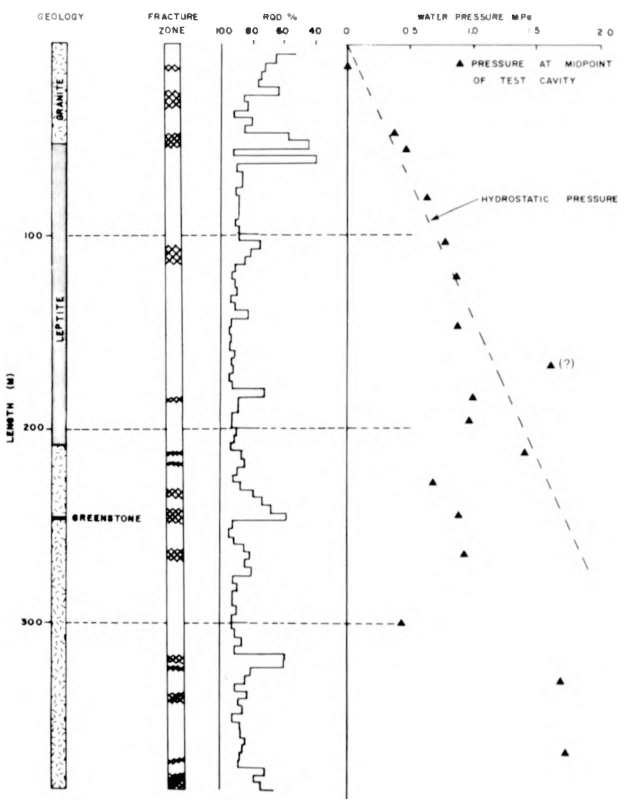

FIGURE 5. GEOLOGY, RQD (ROCK QUALITY DESIGNATION), and WATER-PRESSURE DATA FOR SURFACE HOLE SBH-1

See Figure 1 for location.

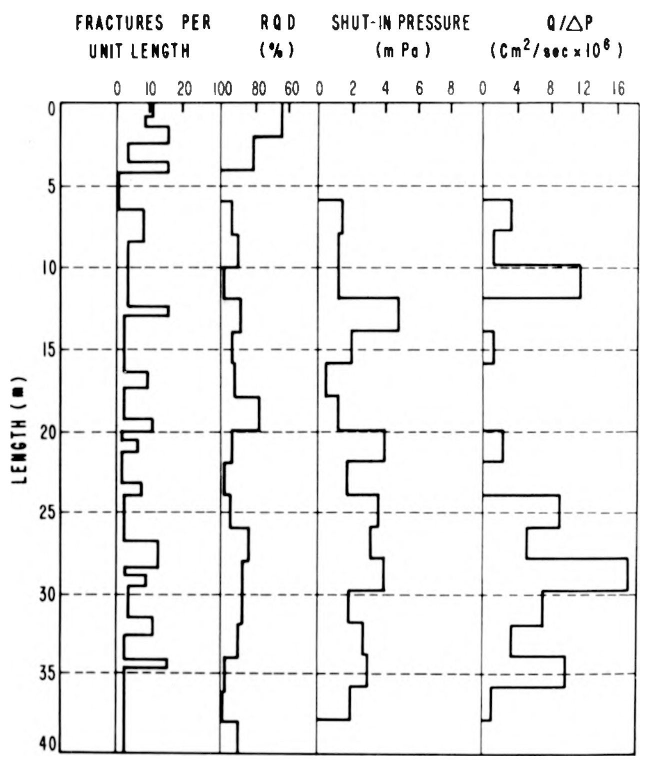

FIGURE 4. RESULTS OF FRACTURE-HYDROLOGY MEASUREMENTS IN S-1[1]

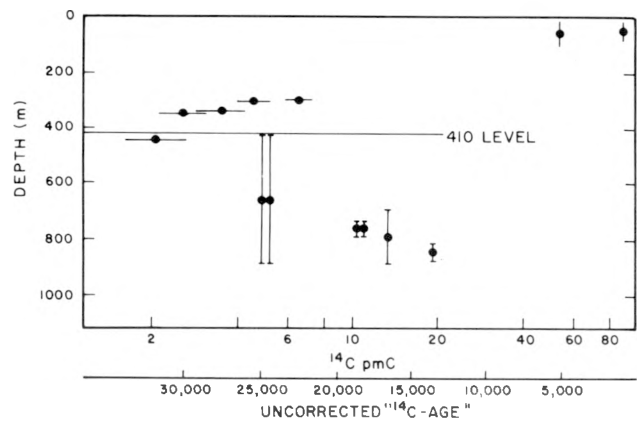

FIGURE 6. ^{14}C AGE DATES FOR GROUNDWATER SAMPLES AS A FUNCTION OF DEPTH

Vertical bars reflect depth ranges over which samples are collected; lateral bars show range of dates obtained.

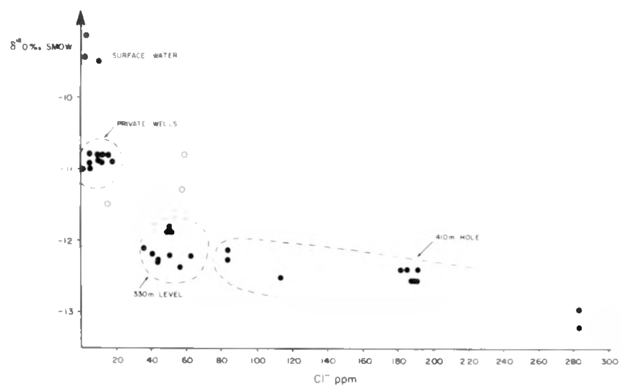

FIGURE 7. PLOT OF $\delta^{18}O$ VERSUS CHLORIDE CONCENTRATION FOR WATERS IN THE STRIPA AREA

Samples of shallow wells (private wells), intermediate wells (330-m level), and deeper (410-m level) groundwaters are outlined.

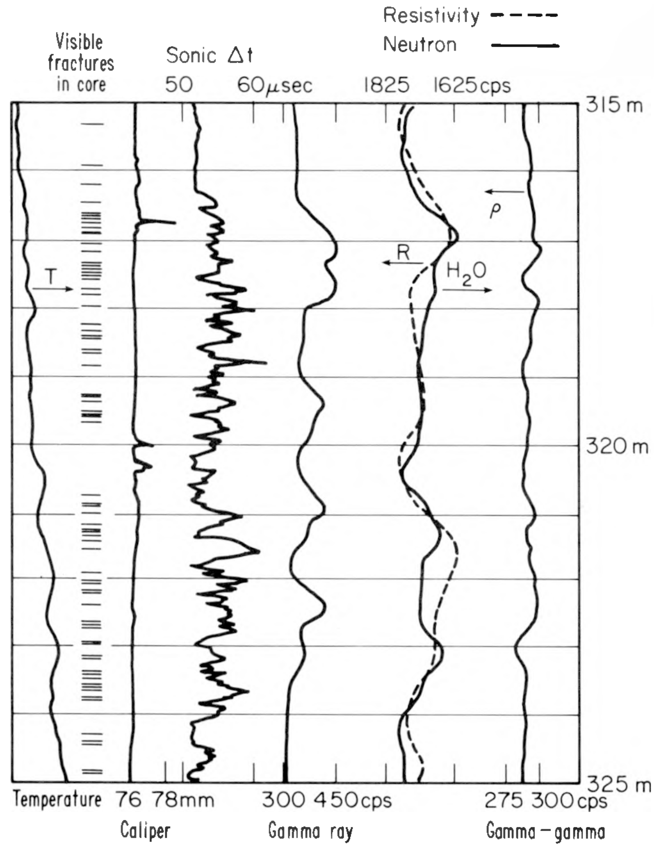

FIGURE 8. RESULTS OF TEMPERATURE, SONIC, GAMMA-RAY, GAMMA-GAMMA, NEUTRON, CALIPER, AND RESISTIVITY LOGS FOR A FRACTURED PORTION OF SBH-1

See Figure 1 for location.

MINING TECHNOLOGY DEVELOPMENT IN CRYSTALLINE ROCK

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Colorado School of Mines

Introduction

A test facility is being established at the Colorado School of Mines Experimental Mine located at Idaho Springs, Colorado, for the purpose of developing rock-excavation procedures and site-selection/evaluation methodology which may be applied for siting and constructing nuclear waste repositories in crystalline rock. As part of the program, a test room 15 ft wide, 10 ft high, and 100 ft long is being driven in the Idaho Springs formation (a granitic gneiss). The details of some of the experiments that will be carried out both during and after completion of the test room are described in the following paragraphs. The room will also provide the site for the thermomechanical test that will be conducted by TerraTek (described in the next paper in this session). Initially, most of the described research will be carried out by graduate students in the mining and geology departments of the Colorado School of Mines. In the future, some programs may be more appropriately done by outside organizations. The program was begun on July 1, 1979, and completion of the room is expected by September 30, 1979.

Mining Technology

The integrity of the rock mass in which the waste containers are to be placed is of the highest importance in the site-selection process, in the construction phase, and then over the lifetime of the waste. It is important that damage to the host rock be minimized during the excavation process. This includes the tunnel roof, walls, and floor. Smooth-wall or pre-split drifting techniques were initiated and perfected in Sweden. They are, in general, poorly developed in this country. In addition, past efforts even in Sweden, have been aimed primarily at minimizing damage to the roof and walls with no special precautions having been used for the floor or lifter holes. A field research program has been undertaken to develop smooth-wall blasting techniques and principles in hard rock which would be applicable to the excavation of canister boreholes and other drifts.

In driving the room, a number of different hole placements and explosive types are being tried. Each of the designs and the design procedures followed are carefully documented, as are the visual results. From this test, blasting procedures and best designs will be recommended.

As part of the excavation procedure, extensometers are being installed in the roof. These, together with surface-mounted convergence points, will provide an indication of rock-mass behavior during the excavation process. Measurements of peak particle velocity and acceleration are also being made.

Initially, it will be possible to make only visual observations of the integrity of the rock around the opening and thereby estimate the effectiveness of the particular blasting pattern. Upon completion of the excavation, the following techniques will be used to assess the extent of the damage zone:

(1) Cross-hole and uphole seismic (TerraTek instrumentation)
(2) The CSM cell modulus device
(3) The Goodman jack modulus device
(4) Examination of drill cores
(5) Examination of the borehole wall by the use of a petroscope.

These procedures will be carried out in both diamond-drilled and percussion-drilled boreholes in the roof, walls, and floor of the opening to a depth greater than the expected damage zone in each of the sections in which a different blasting pattern has been used. Measurements/observations will be made in the roof, floor, and ribs. In this way, one will be able to (1) compare techniques for observing such zones, (2) provide a quantification of blasting damage, and (3) allow an evaluation of geologic effects on blast effectiveness and damage.

Geology

Because of the great influence on permeability of fractures along structural discontinuities such as faults, joints, and bedding, blasting-induced or -aggravated fractures, or openings on the selection of sites, it is of great importance to have methods for (1) identifying and categorizing these features, and (2) predicting the extent and continuity of these features.

Suggestions have been made (Gain, 1978) to improve the collection of structural data from boreholes. This is needed and has a high priority because , at least initially, boreholes will be the only access to the potential site, and decisions will have to be made on this basis. Normally, however, the holes would be quite far apart, and it would be desirable to keep the number relatively small due to potential water-influx problems. The problem of attributing an "extent dimension" to the observations exists. Hence, any one structure observed could, if it were continuous, eliminate consideration of the site. On the other hand, it may have a very limited extent. Therefore, it is very important to have a generic structural/geological data base which would be useful in evaluating such borehole observations. The best and least expensive possibility for establishing such a data base is through the detailed examination of present structures, such as the CSM mine.

This borehole information will provide procedures for eliminating certain generic structures and/or suggest the frequency of sampling (hole spacing, etc.) necessary for description. As part of the research program, a very careful structural mapping of the CSM mine will be conducted using the various openings to observe and map the major and minor stuctures. It is small enough in plan area (approximately 1000 x 600 ft, with a vertical exposure of about 350 ft) and has numerous openings which could be mapped so that it would be manageable, yet provide for development of principles and techniques which could serve as a model for studies made in other formations. This mapping is presently concentrated in the test room and the near vicinity.

The permeability of the fractures surrounding an opening and the overall permeability of the rock mass are extremely important in assessing the suitability of any repository site. At present, the field techniques available for evaluating these generally small permeabilities are very poorly developed. Generally, a relatively long length of a hole is sealed off and the average permeability determined. The resulting permeability measurement may be due to the presence of only one fracture in the whole interval. The excavated room will provide an excellent opportunity for the testing of techniques/tools developed to measure fracture permeability. Because the fracture frequency changes, the apertures will change markedly as one goes from the wall (blast damaged) of the drift into the virgin (undisturbed) host rock.

A considerable amount of geologic and mechanical data is available regarding the rock at those sites to aid in correlation. An integrated program of fracture-permeability equipment/procedure development is being initiated with the tools/techniques being applied to the evaluation of permeability variations around the room.

A HEATED FLATJACK TEST IN GRANITIC GNEISS

M. Board, H. Pratt, and M. Voegele
TerraTek

The design of a nuclear waste repository in a geologic medium requires a detailed knowledge of the thermal, mechanical, and hydrological behavior of the host-rock mass. In particular, the effects of varying stress and temperature conditions on the elastic moduli, thermal properties, and joint permeability must be defined. An in situ rock-mechanics test is being conducted in Colorado to determine the thermomechanical response and fluid-flow characteristics of jointed crystalline rock. The test series will provide a meaningful in situ data base of thermal and mechanical rock properties for input to numerical models, and will provide a test facility for the in situ evaluation and calibration of rock instrumentation and measurement techniques.

A cubic block of jointed rock approximately 2 meters on a side will be excavated by mining on the top and line drilling on the sides (Figure 1). A specially designed flatjack system will apply uniaxial or biaxial stresses to the boundaries. A line of vertical heaters will extend across the block and past the boundries to provide a thermal field. Analytical and numerical analyses are being conducted to provide pretest estimates of stress fields and temperature and displacement distributions, both within the block and outside the block boundaries.

Instrumentation will be installed in both the block interior and beyond block boundaries as well as upon the block surface (Figure 2) to monitor: (1) intact-rock and joint displacement, (2) intact-rock strain, (3) stress, (4) temperature, (5) compressional- and shear-wave velocities, and (6) joint permeability. Instrumentation will include: vertical and angled extensometers, short-gage-length DCDT's, vibrating-wire strain gages, and bonded resistance strain gages to measure displacement and strain; USBM gages, IRAD vibrating-wire gages, and CSIRO gages to measure stress changes; and thermocouples to measure temperature. A specially designed crosshole ultrasonic system will be emplaced in boreholes to measure changes in ultrasonic P- and S-wave velocities to evaluate joint behavior and possible thermal-cracking phenomena. Waveforms will also be recorded for further analysis. A packer system has been designed to measure permeability as a function of stress and temperature. Joint permeability will be measured between pairs of holes using water or air. Flow will be measured by either pressure decay or volume flow at constant pressure.

Prior to excavation of the block, a series of laboratory and field tests will be conducted. Mechanical, physical, and thermal properties are currently being measured on core samples in the laboratory as a function of stress and temperature. In situ stress relief will be evaluated by making pre- and post-block-excavation field measurements of surface strain, ultrasonic velocities (P-wave and S-wave), displacement (joint and matrix), and internal stress.

The in situ test series will consist of a matrix of experiments at a variety of loading conditions, paths, and temperatures. The first test series will consist of loading at ambient temperatures to examine deformational and fluid-flow response. Loads will be applied perpendicular and parallel to the strike of the jointing, and will include uniaxial, biaxial, and proportional stress conditions (Table 1). The maximum applied stresses will be on the order of 15 MPa.

A second series of tests will be conducted to examine the thermal and deformational properties of the rock mass under conditions of varying temperature and applied stress. These tests will define the coupling of thermal and stress effects in the range of expected repository conditions. Test temperatures will range from ambient to approximately 100 C, and applied stresses from 0 to approximately 7 MPa (1000 psi), as shown in Figure 3. The tests are designed to furnish information pertinent to the evaluation of available instrumentation as well as to investigate the following relationships:

- Thermal diffusivity of the rock mass as a function of temperature and stress
- Thermal expansion of the rock mass as a function of temperature and confining stress
- Deformational properties as a function of temperature and stress
- Ultrasonic velocities as a function of temperature and stress.

Additional tests at higher loads (to 20 MPa) and temperatures (to +200 C) are possible; these extreme conditions will be implemented only if they are needed to clearly define rock-mass response, because their use will almost certainly result in a change of the basic rock-mass properties.

TABLE 1. SUMMARY OF LOAD PATHS IN STATIC DEFORMATION TESTS

Load Path	**Stress, MPa**	
	EW	NS
Uniaxial Stress, EW	3.5, 7, 14	0
Uniaxial Stress, EW Reload	3.5, 7, 14	0
Uniaxial Stress, NS	0	3.5, 7, 14
Unaxial Stress, NS Reload	0	3.5, 7, 14
Biaxial Stress, EW/NS=1	3.5, 7, 14	3.5, 7, 14
Biaxial Stress, EW/NS=1, Reload	3.5, 7, 14	3.5, 7, 14
Biaxial Proportional Stress, EW/NS=1	3.5, 7, 14	1.75, 3.5, 7

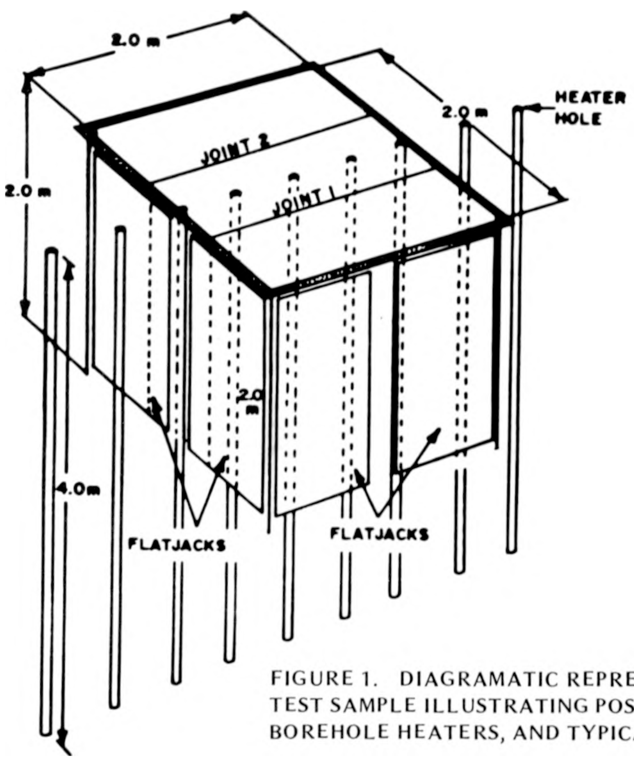

FIGURE 1. DIAGRAMATIC REPRESENTATION OF CUBIC TEST SAMPLE ILLUSTRATING POSITION OF FLATJACKS, BOREHOLE HEATERS, AND TYPICAL JOINTS

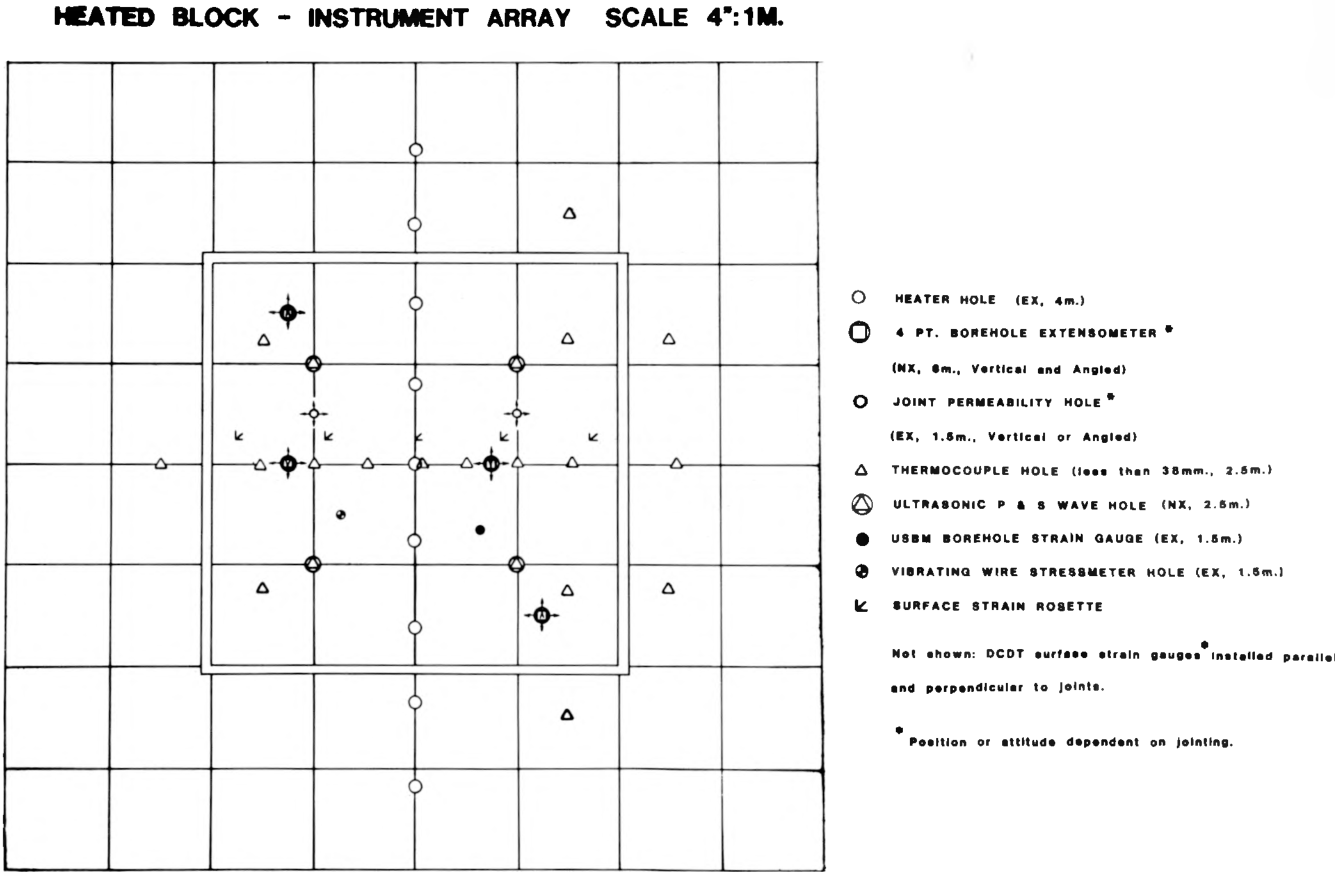

FIGURE 2. PLAN VIEW OF HEATER BLOCK TEST SAMPLE ILLUSTRATING POSITION AND TYPE OF INSTRUMENTATION IMPLEMENTED

T (°C) \ σ^* (MPa)	0	3.5	7
20 (ambient)	X	X	X
50	X	X	X
85	X	X	X

For Each Test Determine:

- Thermal Diffusivity Upon Heating and Cooling
- Thermal Conductivity From Quasi-Steady State
- Thermal Expansion
- Elastic Properties of Intact & Jointed Mass
- Deformation Behavior of Joints
- Ultrasonic Seismic Velocity
- Temperature, Stress and Displacement Fields

*Boundary Stresses Applied Biaxially

FIGURE 3. MATRIX FOR TEMPERATURE/STRESS TESTS

CONASAUGA NEAR-SURFACE HEATER EXPERIMENT: RESULTS AND IMPLICATIONS

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Sandia Laboratories

The Conasauga near-surface test is one of two heater tests that have recently been run at shallow depths in argillaceous materials. The specified objective of the test is to determine whether the heat from a high-level waste canister would alone be sufficient to cause major changes in the geologic environment. When integrated with data from other heater tests in rock salt, basalt, granite, and tuff, a body of literature will result that will contribute to assessing the relative merits of siting a high-level waste repository in each of the various lithologic choices.

The formation chosen for this particular experiment consists predominantly of a friable silty illitic shale. Bedding is pronounced, and the shale is interspersed with numerous strata of limestone and siltstone. The rock has been sheared in a variety of directions as a consequence of the tectonic activity responsible for the Appalachian Mountains. Because rainfall in the region averages 1.7 m per year, water is continuously available to refill the network of hairline fractures that pervade the formation.

The experimental setup is illustrated graphically in Figure 1. Around each of the 3-m-long by 0.3-m-diameter heaters is an array of thermocouples, horizontal stress gages, and vertical extensometers emplaced at various depths in drill holes. Both heaters were operated with a midplane temperature of 385 C. At Site 2 the ingress of water was precluded by applying a 0.19-MPa air pressure to the heater hole. At Site 1 only a 0.08-MPa air pressure was applied. During much of the experiment the bottom few centimeters of the heater were therefore submerged.

Laboratory measurements on Conasauga shale samples suggested that certain events might be anticipated in the heater tests. Thermal-expansion measurements made on unconfined Conasauga samples commonly show a marked initial contraction, due to dehydration of the clays (Figure 2), followed by a slow expansion as the sample is further heated. This could cause a decrease in thermal conductivity due to opening of hairline fractures (Figure 3). Prior to the test, it was postulated that this sort of behavior would occur adjacent to the hottest portion of the heater, seriously interfering with heat dissipation into the formation and greatly increasing its permeability.

In actuality, the consequences of heating were considerably less spectacular. Post-test examination of the heater hole walls showed that, in fact, no significant fracturing or spalling had occurred. Core samples recovered following cool-down confirmed that fracturing had also not occurred in the formation away from the heater. In fact, post-test transmissivity measurements showed that both sites exhibited a decreased permeability as a consequence of heating. Finally, the vertical extensometers responded in a manner characteristic of a slight positive thermal expansion rather than the behavior illustrated in Figure 2.

The lack of major in situ change is reflected in the formation's thermal response. In contrast to the laboratory thermal-conductivity values, the formation dissipated heat as if it had an average thermal conductivity of about 1.8 W/m C. Specific information on the high-temperature region may be gained from the temperature gradient within the 100 C isotherms. This is the region where dehydration could have caused the conductivity to drop precipitously. However, when the heater power levels and the essentially fixed position of the isotherms during the last 210 days of the test are considered, it is apparent that the conductivity must have remained in the range of 1.5 to 1.75 W/m C in this region.

A second point of interest regarding heat transfer concerns the effects of groundwater. For the first 100 days of the test, all of the significant temperature increases occurred within the thermocouple array; consequently, it was possible to show that through-flowing ground water did not remove a detectable amount of thermal energy from the site during this time span. At temperatures below 100 C there is close spatial agreement between isotherms predicted assuming conduction to be the only mode of heat transfer, and those actually measured in the field. In this region, then, it follows that convection is not having a major effect on heat transfer. At higher temperatures along the sides of the heater, the isotherms were essentially vertical. Heat transfer is, therefore, essentially radial, and convecting steam did not remove a significant amount of energy from this area. Above and below the heater, however, there is a consistent upward displacement of the isotherms from what would be predicted assuming conduction alone. Boiling water, apparently, provided a heat sink below the heater, and steam exiting into the formation above the heater delivered more energy to this region than would have been possible through conduction alone. Thus, although the effects of convection are perceptible, most of the heat flow into the formation was transferred by the conduction mode.

Although the primary emphasis of the Conasauga near-surface heater experiment was on the thermal shock to the formation, several other observations have bearing on problems to be encountered during repository operation. A considerable amount of information is to be gained in the laboratory by subjecting small samples of potential canister or waste forms to controlled steady-state conditions for long time periods. In the case of Site 2, such tests were sufficiently close to the actual field situation that surprises were absent. At Site 1 a different picture emerged due to the generation of steam in the lower part of the heater hole. In contrast to the innocuous corrosion observed at Site 2, the upper part of the Site 1 heater showed many small blotches of somewhat intensified corrosion, and, for a distance of about a meter above the heater, the Type 304 stainless steel thermocouple leads had been completely corroded. Below water level a layer of scale was deposited.

Recurrent boiling of groundwater also seemingly influenced the alteration of a sample of PNL 75-25 borosilicate glass waste form simulant that was included in the field test. Depending on the availability of steam or condensate, the sample was alternately dried or wetted as well as being heated or allowed to cool. At the conclusion of the test, a gel layer had formed that was up to 20 microns thick. A network of shrinkage cracks had formed due to dehydration on the internal surface of the gel layer, and this texture was reflected in the scalloped surface of the underlying unaltered glass. Thus, this behavior suggests that assessments of the durability of a borosilicate glass waste form might include the study of the effects of alternate wetting and drying as well as the rate of initial gel-layer formation.

To conclude, tests such as the Conasauga near-surface heater experiment will not resolve all the problems related to the geologic disposal of nuclear waste. The effects of overburden pressure and radiation are yet to be assessed even in the most cursory manner. Further, even within a single rock designation such as shale, the possible variation in properties is so great that a vast amount of work will be required before a comprehensive understanding of even this one lithology in the near-surface environment will result. In light of the present level of understanding, it is not surprising that, on occasion, the observed behavior was not anticipated prior to the test. It is, however, encouraging to note that, in spite of an occasional surprise, nothing to date has been found which would preclude the use of argillaceous rocks for the disposal of high-level nuclear waste.

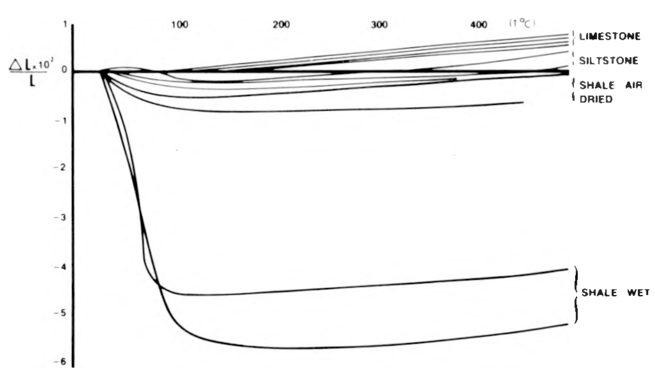

FIGURE 2. THERMAL EXPANSION OF CONASAUGA SAMPLES

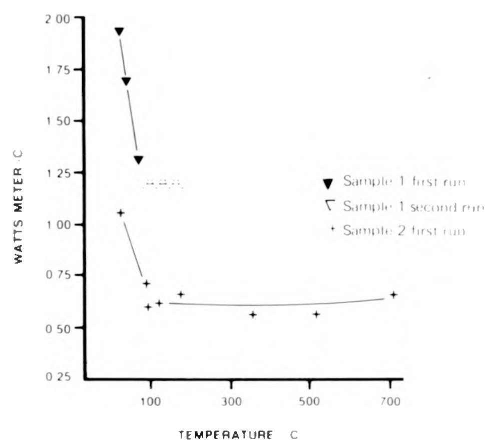

FIGURE 3. THERMAL CONDUCTIVITY OF CONASAUGA SAMPLES

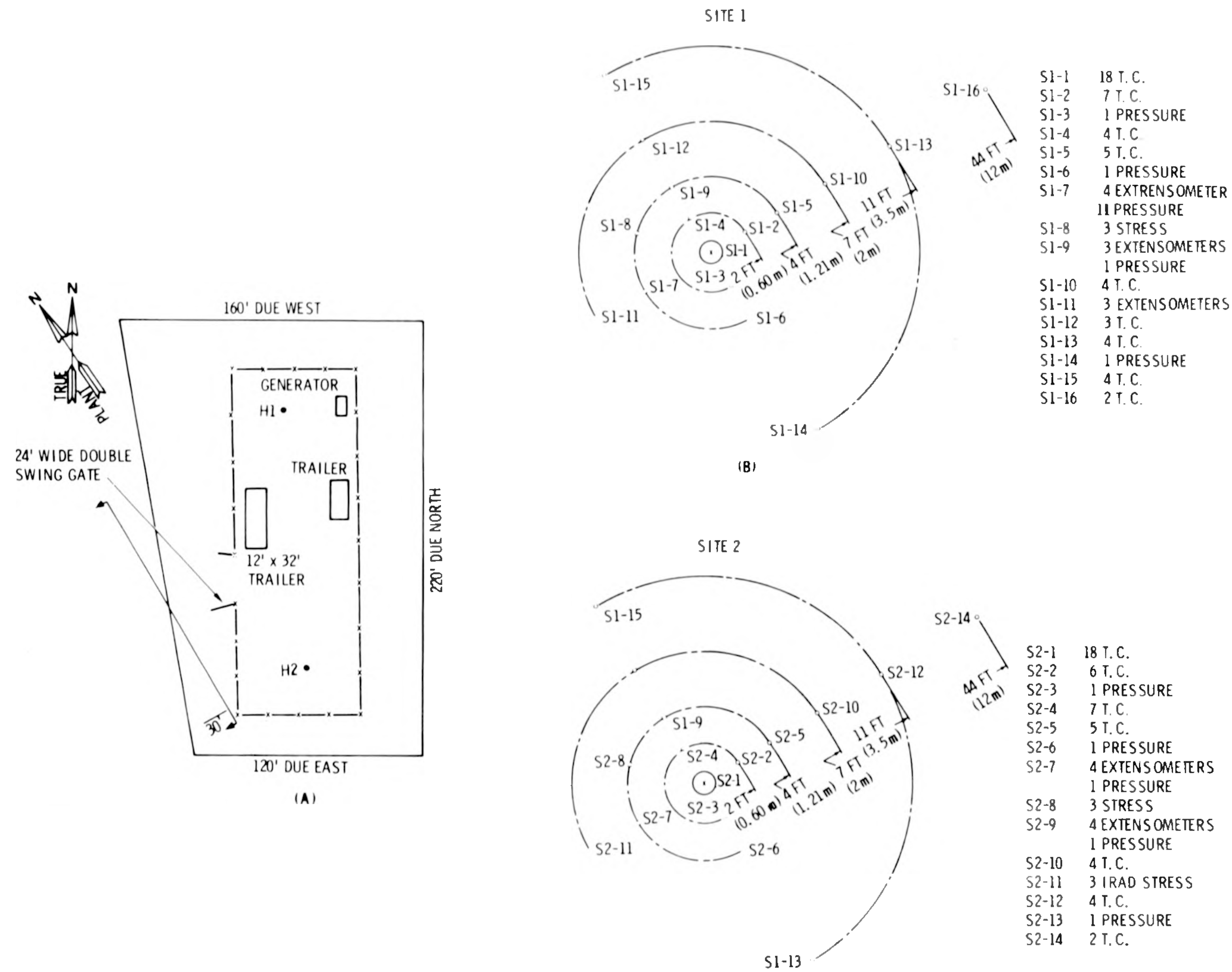

FIGURE 1. SITE LAYOUT CONASAUGA NEAR-SURFACE EXPERIMENT

PLANNING OF SALT TEST FACILITY

R. W. Brown
Office of Nuclear Waste Isolation

A Salt Test Facility (STF) is currently in the planning stage. It is envisioned to be both a scientific test area or a laboratory, and an operational testing arena. Current plans are that many tests will be conducted simultaneously in both of these areas. Past, ongoing, and future tests at many different locations either have used or will use modeling techniques that simulate the effects of nuclear waste. Testing at the STF will include activities that involve radionuclides. As shown in Table 1, the STF will provide information for use in licensing and design and will thereby reduce the dependency on extrapolation of laboratory data and expert judgment. In situ testing goals are model validation and reduction of uncertainty, determination of failure limits, and resolution of specific issues involving radioactivity effects. Data from these tests will allow correlation of the various effects measured in off-site laboratories and in situ tests.

Collocated with the scientific test area in STF will be an operational-type test area in which machines and operational methods can be evaluated in an environment quite similar to a repository. Tests to validate concepts and operational effectiveness of waste-placement and -retrieval equipment and methods are anticipated, and demonstration of monitoring systems will also be accomplished. Tests which will measure the efficiency of alternative mining and retrieval systems may be conducted.

The underground mine should provide areas in which testing and supporting functions enumerated in Table 2 can be accomplished. These areas will be adjacent one to the other and advanced planning is being conducted to prevent one test from affecting the results of another.

Table 3 enumerates the facility design criteria—what we anticipate the STF to be. It is planned that the facility will be made available to all NWTS program elements and to outside groups for tests with experimental objectives relevant to waste-disposal issues. It is expected that the facility will have a 15-year design lifetime. Licensing of the facility is not expected because it will be an R&D facility in which the waste materials will be completely removed at the end of the tests and will be nominal in quantity, perhaps 20 to 30 canisters. The site is planned to be located in an existing mine that is no longer in operation or in a mine in which a portion is no longer being mined. The mine will be located between 800 and 2000 ft below the surface. Renovation will provide for personnel safety and for safe transfer of the radioactive material. Use will be made of existing electrical, transportation, and ventilation systems, as much as is possible. All operations will be subject to quality controls specified by ANSI N45.2

Figure 1 depicts the STF milestones based on receipt of approval from DOE to proceed at an early but as yet unknown date.

Table 4 shows progress to date. Basic screening of existing openings (or mines) in salt has been accomplished. This screening ranked such openings according to availability, physical acceptability, geologic environment, and other environmental considerations. Also, preliminary contractual documents have been prepared and functional design criteria which will support the basic contractual documents are now being completed.

Finally, in terms of future progress, the activities enumerated in Table 5 will be completed in the next few months. As can be seen, our intention is to inform public bodies of our proposed activities before we attempt negotiations with the owners or operators of potential candiate sites.

In conclusion, we believe that the STF is essential to the effective and efficient acquisition of data needed for the design of repositories in salt. The availability of the STF for conducting in situ testing work, we feel, will move NWTS/ONWI much closer to the ultimate goal of providing permanent isolation of nuclear waste from the biosphere.

TABLE 1. OUTPUT AND GOALS FOR SALT TEST FACILITY

In Situ Testing:

(1) Provides hard data for use in design and verification of safety performance assessment computer models

(2) Minimizes extrapolation and expert judgment to predict outcome

(3) Accomplishes three goals:

- Model validation and reduction of uncertainty
- Determination of failure limits
- Resolution of specific technical issues involving radioactivity effects

TABLE 2. SALT TEST FACILITY UNDERGROUND AREAS

- Simulation testing with nonradioactive materials
- Radionuclide testing
- Operational demonstration of equipment
- Test and facility support functions

TABLE 3. DESIGN CRITERIA FOR SALT TEST FACILITY

- Possible 15-year life
- Licensing of facility not anticipated
- Small quantities of radioactive material
- Use of existing mine opening (closed or not used)
- 800 to 2000 feet below surface
- Renovate as required for complete safety
- Maximize use of existing mine supporting systems
- Decontamination and decommissioning of radioactive test sites
- Quality assurance – ANSI N45.2 (1977 ed.)

TABLE 4. PROGRESS TO DATE ON SALT TEST FACILITY

Site Screening and Selection:

- Basic screening completed
  - Literature review
  - Criteria defined
    - Availability of mine
    - Geologic environment
    - Other environmental considerations
- List of candidate sites provided to DOE

A-E Selection:

- Preliminary contractual document to DOE
- Functional design criteria prepared

TABLE 5. NEAR-TERM ACTIONS IN DEVELOPING A SALT TEST FACILITY

- Final review of site screening criteria and results
- Initiate A-E design contractor selection process
- Begin coordination with state and local agencies
- Finalize functional design criteria

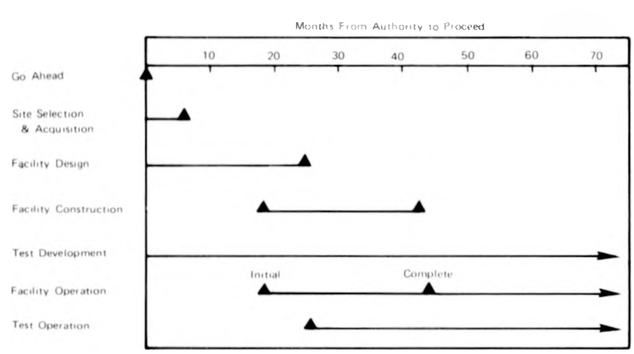

FIGURE 1. SALT TEST FACILITY PROJECT SCHEDULE

REPOSITORY EQUIPMENT DEVELOPMENT METHODOLOGY

OVERVIEW OF ENGINEERING DESIGN-CRITERIA METHODOLOGY DEVELOPMENT

R. A. Cudnik
Office of Nuclear Waste Isolation

Generic design criteria, including requirements for repository equipment and testing procedures in NWTS projects, serve two primary functions. First, they provide a basis for preliminary and detailed design and the establishment of testing procedures. Second, they provide a basis for evaluation of design efforts and impacts and comparison of design alternatives.

What are design criteria and requirements and how do they compare to design specifications? We offer the following definitions:

Design Criteria: These are standards or patterns for guidance in making design judgements. Criteria should have a sound basis or rationale and should not cover the level of detail provided in specifications.

Design Requirements: These are the needs or conditions which must be satisfied by a design.

Design Specifications: These are detailed descriptions setting forth the limits or particulars such as dimensions or materials for engineering work. Specifications usually provide numerical values and are developed from criteria and requirements.

Engineering design criteria and requirements thus are the standards to which designs are developed and from which specifications, the detailed descriptions setting forth design particulars, are developed.

Before equipment design criteria and requirements can be applied in the design effort, they must be established. To achieve this, and to assure that the criteria reflect the objectives, constraints, and requirements imposed by the overall repository system, NWTS/ONWI is developing a systematic methodology for the establishment of engineering design criteria.

Figure 1 provides an overview of the design process and the role of the design-criteria methodology, including the functions it provides within the design process. As shown in the diagram, the methodology takes as input the constraints and requirements imposed by the NWTS Program objectives and requirements, the repository conceptual designs (or detailed designs as they evolve), and applicable regulations, codes, and standards. The methodology provides as output design criteria used in the receiving, storage, transfer, emplacement, and retrieval functions dictated by the nuclear waste-isolation concept under consideration.

The methodology provides two main functions to the design process. The first involves establishing the equipment design basis, controlling the design and its modifications as it evolves, verifying the design, and establishing the basis for equipment testing and subsequent acquisition. The second function involves analyzing all interfaces for impacts and assessing the impacts of equipment design modifications on other repository elements.

The development of a methodology for establishing engineering design criteria and requirements for repository equipment will encompass four steps:

(1) In-depth review of NWTS Program documents and existing repository conceptual-design studies and subsequent revisions to identify repository characteristics that bear on equipment design criteria and requirements

(2) Assembly of an overall repository functional-flow diagram based on repository system and functional characteristics

(3) Development of a methodology providing a systematic and thorough evaluation technique for establishing equipment design criteria and subsequent design control, verification, and equipment acquisition

(4) Organization of the methodology such that it can be readily used to establish equipment design criteria as well as serve as a checklist to verify that existing or newly designed equipment meets established design criteria.

As part of the overall design-criteria methodology development effort, it is further planned to develop an approach to apply reliability and maintainability analysis techniques to the availability requirements of a nuclear waste repository. The objective here is to establish reliability and maintainability criteria to be used in the design of repository equipment to assure that the overall repository availability requirements can be satisfied.

In summary, the methodology development efforts are directed at assuring that the design criteria and requirements established for repository equipment and testing procedures have as their basis the overall NWTS Program objectives and requirements and the repository-system requirements and constraints. The work will promote subsequent development of design and engineering criteria and specifications for test and operational procedures and equipment that will smoothly function in and interface with the overall repository system.

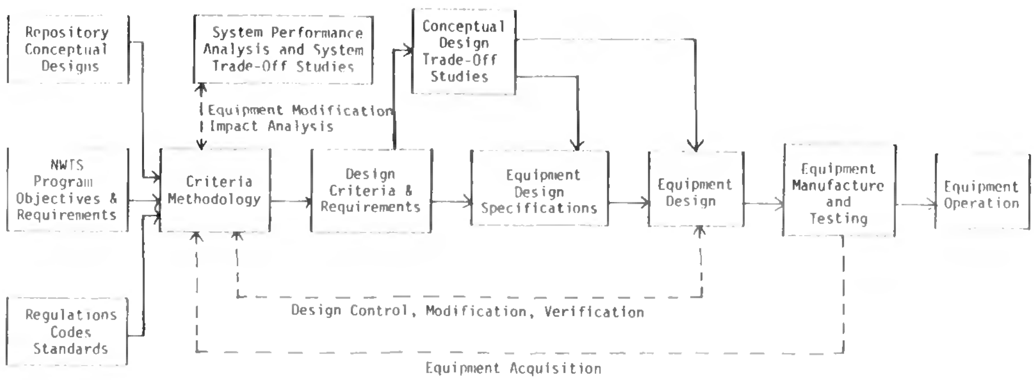

FIGURE 1: OVERVIEW OF THE DESIGN PROCESS FOR REPOSITORY EQUIPMENT

AN APPROACH TO EQUIPMENT CRITERIA AND SPECIFICATION DEVELOPMENT WITH CANISTER EVALUATION AS AN EXAMPLE

R. Wilems, J. Arbital, and S. Grady
Science Applications, Inc.

The development of repository-equipment criteria and specifications requires the consideration of the integration of the equipment into the total system, including the man-made and natural environment in which the equipment must operate. The objective of this project was to identify a methodology appropriate for evaluating the expected performance of various waste-canister concepts within their emplacement environments. The methodology was to be generally applicable to the establishment of equipment criteria and hardware specifications from systems-level facility and geologic criteria. In order to demonstrate the application of the methodology, two canister concepts defined by ONWI were to be evaluated.

The methodology[1] developed under this contract is based on classical systems-engineering techniques. The basic steps involved in the methodology include: (1) the development of functional-flow diagrams at the appropriate level of detail (equipment level), (2) identification of functional and interface requirements at the equipment level, (3) analysis of the requirements to identify key equipment design parameters, (4) identification of appropriate measures of acceptability, and (5) use of a modified DELPHI technique to evaluate candidate or proposed equipment.

From various preconceptual[2] and conceptual[3,4] repository designs, functional-flow diagrams were developed for the waste-management processes that involve the waste canister. Four major processes were identified: waste packaging and transportation, repository surface-facility handling, waste emplacement, and waste retrieval. The functional-flow diagrams developed for each of these processes are shown in Figures 1 through 4. An analysis of these functional-flow diagrams led to establishment of requirements for each of the functions represented. For example, for the process of waste packaging and transportation a function might be to seal the canister and inspect the seal at the packaging plant. A requirement associated with this function might be that the canister material must be capable of being bonded to a lid of similar material and the material adjacent to the seal must maintain integrity.

Associated with the requirements for each function, key canister parameters were identified; for example, for the sealing and inspection function mentioned above, the canister material must have an acceptable bondability, coefficient of thermal expansion and strength. Following this process, the material and mechanical parameters listed in Table 1 were identified as keys for evaluating candidate canister designs.

The limited scope and resources of this project did not allow the problem of optimum canister design to be addressed. If this had been the objective of the study, the systems-engineering methodology could have been used to specify an appropriate canister or canisters. The objective of this project, however, was to demonstrate the applicability of such an approach through the evaluation of two specific canister designs. The designs were checked against the critical parameters derived from the functional-flow analysis to illustrate the evolution necessary to develop specifications. In this project, time and resources did not permit quantification of the specifications which would be the objective of an actual application of the methodology; however, qualitative comparisons of the candidates were accomplished.

The candidate canisters identified for evaluation by ONWI were a bare, single-barrier, stainless steel canister and a multibarrier canister consisting of a stainless steel tube encapsulated within an aluminum oxide capsule. The single-barrier canister is similar to the various stainless steel canisters employed in the different U.S. preconceptual and conceptual repository designs whereas the multibarrier canister is one of the candidates investigated in Sweden and documented in KBS reports.[5,6]

The comparison of each canister's specifications and supporting technical documentation showed that either canister would probably meet the various requirements with the following reservations:

(1) The encapsulated canister has adequate strength to withstand normal handling procedures, but the ceramic (Al_2O_3) is generally brittle and its resistance to a drop onto a hard surface is not apparent. An experiment should be conducted before concluding that this requirement is met.

(2) Adequate handling lugs would have to be adapted to the casing enclosing the encapsulated canister.

(3) If corrosion resistance is defined as resistance to degradation of canister wall thickness through chemical action, the bare stainless steel canister would be adequate for short periods of time, but it may not be adequate if an extended period of integrity is desired. On the other hand, if stress corrosion cracking is of concern, the bare canister may well experience this type of degradation even shortly after emplacement.

(4) The heavier weight of the encapsulated canister appears tolerable, but hoist requirements should be checked.

(5) The chemical inertness of the bare stainless steel canister is uncertain during extended storage periods.

(6) The encapsulated canister may require larger manufacturing tolerances than the bare canister; however, these should pose no problems.

(7) The high-density aluminum oxide surface of the encapsulated canister probably has a surface that can be decontaminated, but this should be verified.

Once the candidate canisters have been proven to meet requirements associated with the various key design parameters, other factors should be considered in the selection of an optimal design. Some obvious considerations include: cost, availability of materials, R D & D requirements, and licensability. One technique for evaluation with respect to these factors is a modified DELPHI technique. At the heart of this technique is a decision matrix composed of rows representing alternative designs and columns representing evaluation factors. The technique combines analyses based on fact and expert opinion to assign appropriate weights to the pertinent evaluation factors and to specify relative values for each entry in the matrix.

It is recommended that evaluation factors that are easily quantified be used wherever possible. For factors that are not easily quantified the procedure suggested is an artificial quantification, using an expert panel. This quantification might be accomplished by having each panel member assign the top value in the scale to the alternative he considers best and then assign values to the remaining alternatives in proportion to the panel member's subjective evaluation of those alternatives compared with the best alternative. Values of alternative components for the dimension could then be obtained by taking the average of the scores assigned to the component by the experts. In this way, parameters that can be quantified by experimental data or calculation can be readily combined with best engineering judgement associated with parameters that are not readily quantified. The weighted combination of these values would then identify the optimal choice.

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Table 1. Key Design Parameters for Canister Evaluation

Material Parameters:

- Strength
- Coefficient of thermal expansion
- Corrosion Resistance
- Density (weight per unit volume)
- Weight
- Porosity
- Tolerance to radiation
- Chemical inertness
- Bondability
- Melting point

Mechanical Parameters:

- Dimensions
- Tolerances of dimensions
- Gripping attachment
- Surface texture
- Manufacturability

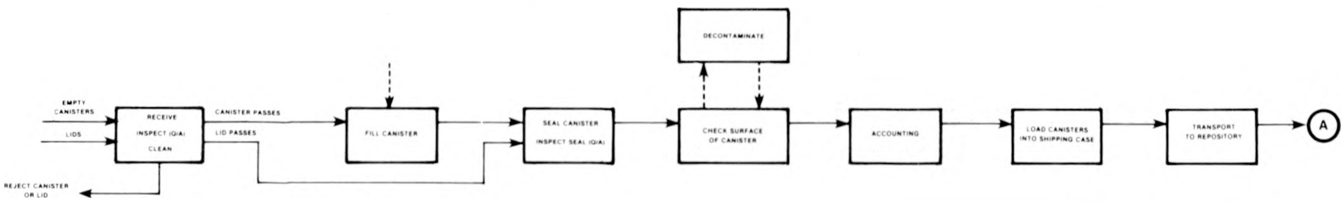

FIGURE 1. PACKAGING PLANT FUNCTIONAL FLOW

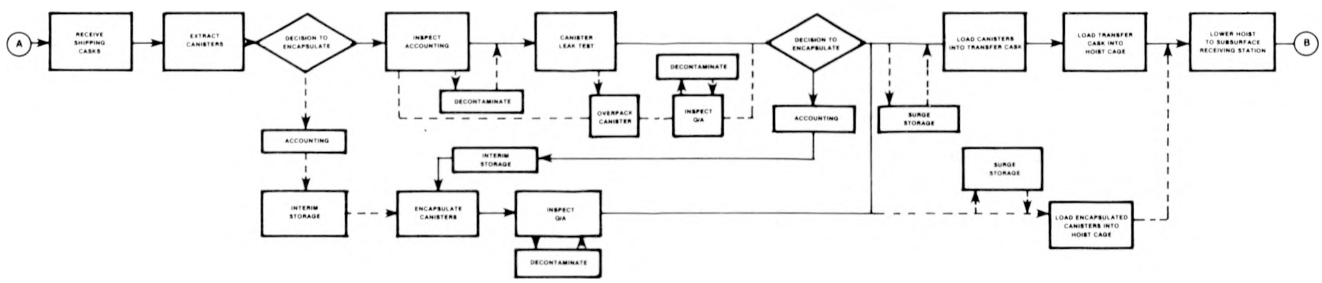

FIGURE 2. REPOSITORY SURFACE-FACILITY FUNCTIONAL FLOW

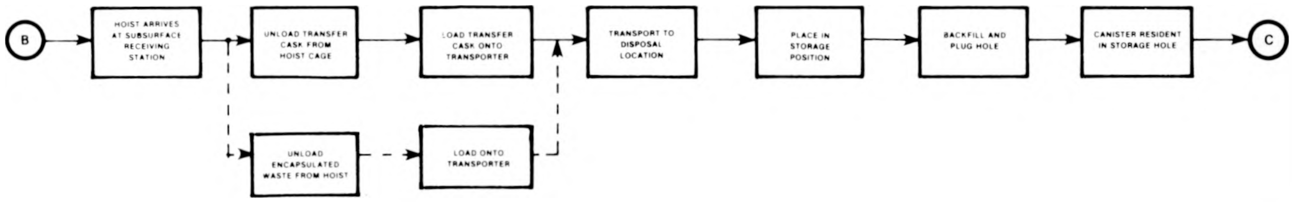

FIGURE 3. WASTE-EMPLACEMENT FUNCTIONAL FLOW

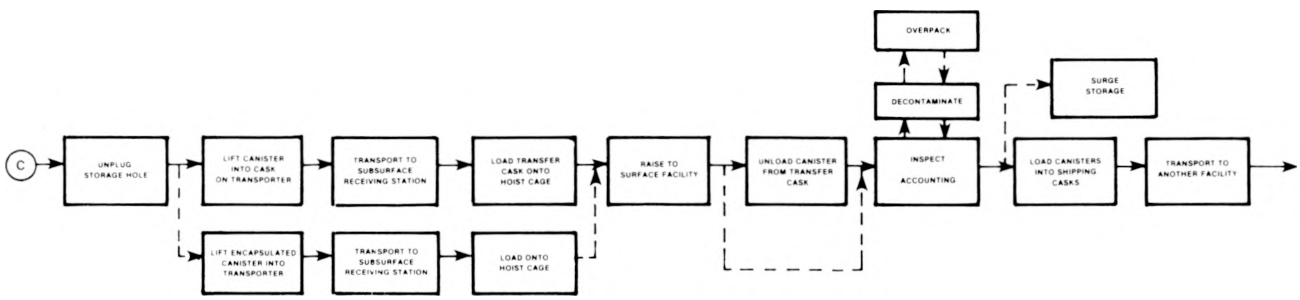

FIGURE 4. WASTE RETRIEVAL FUNCTIONAL FLOW

REPOSITORY-EQUIPMENT DEVELOPMENT OVERVIEW

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Office of Nuclear Waste Isolation

Repository equipment development activities (including instrumentation development) are required whenever new equipment designs must be utilized or shelf equipment must be extensively modified or adapted to meet repository needs. Examples of these specialized needs include equipment adaptation to hostile environments, special reliability or availability constraints, special concerns for operator or public safety, and functional requirements which are unique to the repository concept.

These equipment and instrumentation development activities are, in essence, a service function in the NWTS Program. These services include providing designs and/or design specifications to the repository-facility engineering program for incorporation by the A/E into the repository design. In addition, the NWTS/ONWI technology development program requires equipment as part of the model synthesis and verification efforts. An example of this type of equipment requirement is the instrumentation development work needed to perform many of the in situ experiments under hostile environmental conditions.

Planning Activities for Equipment Development

The planning activities at ONWI for equipment development requirements are focused within two committees: the Repository Equipment Committee and the Instrumentation Equipment Committee. The Repository Equipment Committee's focus is on materials-handling, ventilation, mining, and, to a certain extent, security and safeguard equipment which are part of the repository-facility design. The Instrumentation Committee not only considers the repository-facility instrumentation requirements, but is also cognizant of other instrumentation needs, such as for in situ testing . One major objective of each committee is to recognize the differing requirements which must be satisfied by the equipment to be developed. A corollary objective is to identify differing requirements which are in conflict and either resolve the conflict or require that the conflict be resolved as part of the design effort by the subcontractor.

Another major objective of the committees is to coordinate procurement efforts within ONWI such that prioritized needs are met within the confines of budget and milestone restrictions. The prioritization process is relatively complex in that it involves estimates of the developmental effort involved (time and cost), the probability of success, the benefits to be derived, the status of the facility design, the overall NWTS/ONWI major milestones which must be met, and, of course, dollar limitations. Cognizance of other major nuclear-waste project efforts must also be maintained to avoid duplication of effort and to utilize existing equipment designs and information wherever possible.

It should be obvious that such committee planning activities are intended to be fluid and are inherently iterative. New information alters plans, and equipment development efforts must be altered to fit new plans, directives, and priorities.

Equipment Development Activities

Three presentations will be made which illustrate the technical content of representative studies for equipment and instrumentation development.

The first presentation, by SAI, discusses the methodology and results of an effort to generate specifications for repository emplacement/retrieval equipment on the basis that the retrieval function will be "moderately difficult". SAI will discuss what is meant by moderately difficult retrieval and the relative impact of this scenario.

The second presentation, by IRT Corporation, discusses a wireless telemetry system (currently under preparation for field test) which is designed to transmit near-field repository monitoring data to the biosphere without the potential to generate a pathway from the repository to the earth's surface.

The third presentation is an overview of instrumentation requirements for planned in situ experiments emphasizing hostile environment/ equipment interactions. This study is being conducted by the Ohio State University.

RETRIEVABILITY ANALYSIS AND SPECIFICATION DEVELOPMENT

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Science Applications, Inc.

The objective of this project is to develop a specification envelope for canistered-waste emplacement equipment consistent with a "moderately difficult" retrieval concept. The guidelines given for moderately difficult retrieval were ". . .more difficult than ready retrievability and less difficult than retrieval of a failed waste container. The waste canister would be in one piece though it may be corroded and structural strength is suspect". The concept of moderately difficult retrieval was to be based on, but not limited to the following considerations:

(1) Compatibility with ONWI repository concepts[1-3]
(2) Specific throughput rates
(3) Reliability of equipment
(4) Public health and safety considerations
(5) Ease of maintenance
(6) Impact on repository availability
(7) Development requirements for the machine(s)
(8) Safeguards
(9) Cost.

The concept was to consider commercial high-level waste (HLW), including spent unreprocessed fuel, and defense high-level waste. Further, the concept was to be developed independent of the host media, at least with respect to the functional operations being performed. The development of a moderately difficult retrieval concept required initial coordination between this project and one performed by Kaiser Engineers—"A Retrieval Options Study". In a joint workshop, six basic levels of difficulty of retrieval and three basic levels of recovery were developed. The distinction between retrieval and recovery is that retrieval involves intact containers while recovery involves failed containers and, consequently, contaminated rock. It was determined that levels of retrieval difficulty could be identified on the basis of whether the repository room is open or closed, whether the waste canister is free or tightly wedged in its emplacement position, and whether the canister is known to be structurally sound, fragile, or breached. These classifications of retrieval and recovery are summarized in Table 1. It was determined that the "moderately difficult" retrieval concept would be structured about Retrieval Class D because this class involved the retrieval of fragile canisters from a difficult (tight) environment but under expected conditions.

Development of acceptable equipment design criteria and specifications was accomplished through an adaptation of a classical systems engineering approach.[4] For this application, there was a balance to be struck between (1) defining, and thus confining, the criteria and specifications sufficiently well that they will be compatible with the repository itself (not defined in detail) and (2) allowing flexibility to nurture innovation and invention during design and development of the equipment. The process consists of the allocation and development of requirements through the use of a multilevel functional hierarchy of the canistered waste handling within the repository. The elements of this hierarchy appropriate for this project are shown in Figure 1.

For the purposes of this project, the requirements at each functional level were categorized as functional, performance, safety and licensing, and interface. The first three of these types of requirements are generated at a given level through an allocation process from the next higher level, employing an understanding of the role of a specific element at the given level with the element from the higher level. On the other hand, interface requirements are generally evolved through an understanding of the relationships between the various elements at a given level and the relationships between an element and the environment it must operate within. This process defines design analyses that must be performed and design decisions that must be made, but the process places the analyses and decisions into perspective by clearly identifying the rationale for the analyses or decisions and their influence on other aspects of the design. A simplified schematic of the process is shown in Figure 2. Examples of some of the equipment requirements so generated at level 2 are shown in Table 2, while the associated design decisions identified by the design-analysis process are listed in Table 3.

At the time of preparing this project summary, design decisions to be made have been identified through the equipment level (level 3). Appropriate decisions are soon to be made to select options (e.g., via DELPHI process) or identify areas where further analyses are to be performed to assist in selection. Subsequently, within the guidelines of the restrictions generated by the design decisions, the envelope of emplacement-equipment specifications and design criteria will evolve.

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TABLE 1. CANISTER RETRIEVAL AND RECOVERY CLASSIFICATIONS

Operation	Classification	Room	Condition of Canister in Emplacement Position	Canister Structural Status
Retrieval	Class A	Open	Free	Sound
Retrieval	Class B	Open	Free	Fragile
Retrieval	Class C	Open	Design Tight	Sound
Retrieval	Class D	Open	Design Tight	Fragile
Retrieval	Class E	Open	Uncontrolled Tight	Sound
Retrieval	Class F	Open	Uncontrolled Tight	Fragile
Recovery	Class A	Open	Tight	Failed
Recovery	Class B	Closed	Tight	Failed
Recovery	Class C	Decommissioned	Tight	Failed

TABLE 2. EXAMPLES OF EQUIPMENT REQUIREMENTS

Level	Function	Functional Requirements (2.1)	Performance Requirements (2.2)	Safety and Licensing Requirements (2.3)	Interface Requirements (2.4)	Characteristics
2	Canistered waste retrieval	2.11 Retrieve intact waste canister from storage position, transport extracted canister to mine level waste shaft receiving station for transfer to surface	2.21 Retrieve waste canisters and transport to waste shaft for subsequent surface transfer at a rate consistent with the concept of moderately difficult retrieval 2.22 Handle canisters in such a manner as to maintain the integrity of the package throughout the retrieval process and subsurface transfers 2.23 Provide subsystem reliability and maintainability consistent with waste-handling system allocations for maintenance of the overall availability goal during retrieval operations	2.31 Procedures will adhere to the guidelines of the repository radiation control and mine safety programs developed for personnel protection during the retrieval phase of operation	2.41 Interface directly with canister emplacement geometry 2.42 Interface with other subsurface operations utilizing the same mine corridors and rooms 2.43 Interface directly with canistered-waste shaft transfer mechanism	Radiation control limits Mine safety considerations Subsurface logistics plan Canistered-waste shaft transfer mechanism Emplacement geometry

TABLE 3. EXAMPLES OF DESIGN DECISIONS

Functional Level	Requirement I.D.	Design Decision
Level 2		**CANISTERED-WASTE EMPLACEMENT**
	2.11	Define waste-emplacement functions
	2.21	Definition of canister emplacement rate
	2.22	Identify canister handling techniques during emplacement
	2.23	Allocation of reliability goal to waste-emplacement system
	2.31	Identification of radiation control and mine safety program requirements on emplacement operations
	2.41	Identification of waste shaft transfer mechanism
	2.42	Outline of subsurface logistics plan during emplacement
	2.43	Definition of canister emplacement geometry
Level 2		**CANISTERED-WASTE RETRIEVAL**
	2.11	Define waste-retrieval functions
	2.21	Definition of canister retrieval rate
	2.22	Identify canister handling techniques during retrieval
	2.23	Allocation of the reliability goal to the canister retrieval system
	2.31	Identification of radiation control and mine safety program requirements on retrieval operations
	2.41	Definition of emplacement geometry
	2.42	Outline of subsurface logistics plan during retrieval
	2.43	Identification of waste shaft transfer mechanism

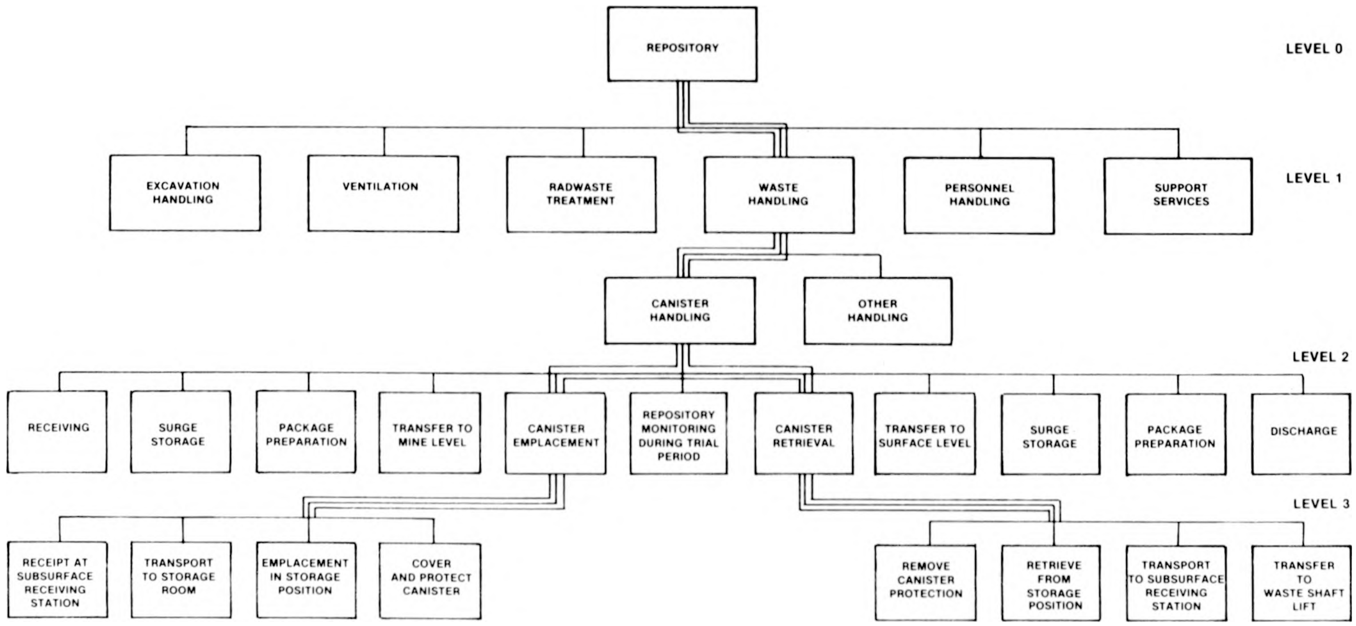

FIGURE 1. REPOSITORY CANISTERED-WASTE HANDLING FUNCTIONAL HIERARCHY

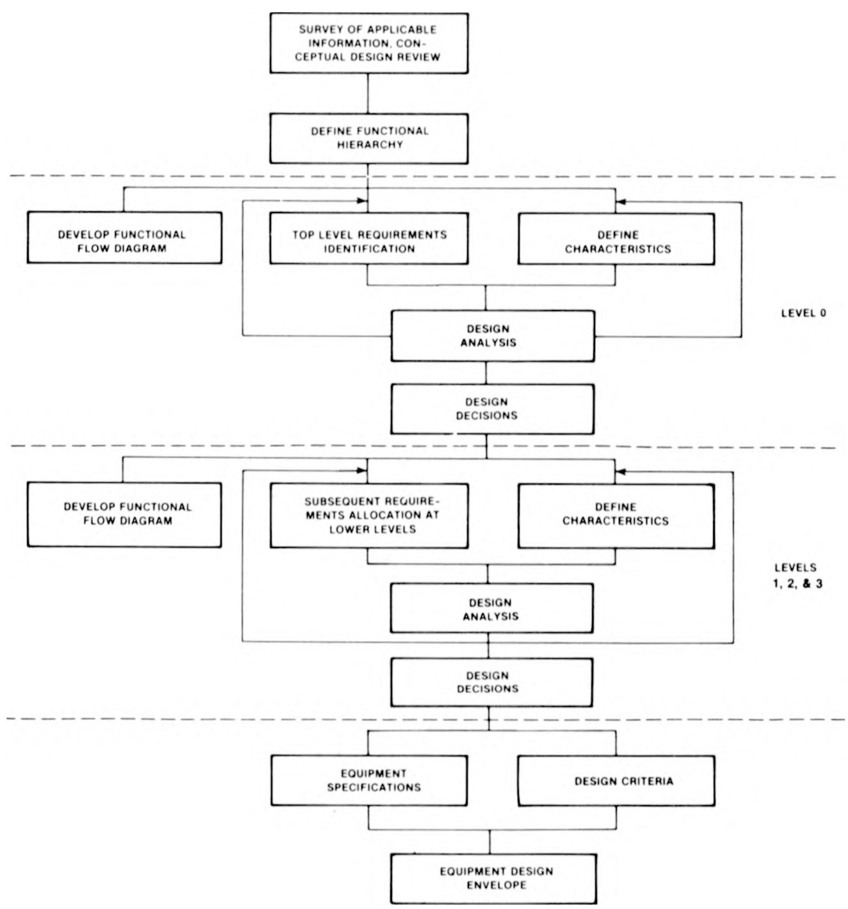

FIGURE 2. REQUIREMENTS ALLOCATION FLOWCHART

TELEMETRY SYSTEM FOR IN SITU MEASUREMENTS

D. Swift, R. Galloway, and J. Stokes
IRT Corporation

The in situ measurement of parameters within a borehole plug can be a valuable means of evaluating and monitoring the performance of such a plug. To preclude possible compromise of plug integrity by the presence of the measurement system, it is desirable that no instrumentation cables, etc., penetrate the plug. The objective of this work is to develop, fabricate, and demonstrate a system which will measure a physical parameter within a plug and telemeter the data to the surface without compromising the plug integrity. Such a system has been developed and fabricated and will be demonstrated in the field during FY 1980. This system consists of an encapsulated battery-powered sensor package designed to be placed within a plug and a set of uphole electronics which controls and communicates with the sensor package via a very-low-frequency electromagnetic-field transmission system. This system has other potential applications in repository monitoring.

System Configuration

The system consists of a set of uphole (wellhead) electronics and associated antenna and an underground (sensor) package as shown in Figure 1. The wellhead electronics package is shown in Figure 2. The underground package, which is shown in Figure 3, includes two encapsulated 6-inch-diameter canisters containing downhole telemetry electronics, batteries, and antenna (17 feet long) and the sensor package with data-acquisition and -processing computer (8 feet long). These canisters are interconnected mechanically as shown and electrically by an armored 12-conductor cable.

The Sensor

For the present program it was decided to focus on moisture detection as the best all-around leak-detection principle that is embodied in a single off-the-shelf device. The choice of moisture detection was based on the fact that leaks into or out of the storage cavity are likely to include water as the major constituent of the leaking material. This prescription obviously does not cover all situations, e.g., radioactive gas leaks, the appearance of a leakage path before leakage takes place, etc. There does not appear to be one sensor capable of handling all possible situations. Any sensor package capable of seeing all types of leaks, however, will no doubt include a water detector as one of the key sensors. Also, a ground rule for this sensor was that it be a commercial off-the-shelf tool so that the development effort would be focused on the telemetry system. For these reasons, moisture detection was a natural choice for the sensor to be used in the first experimental borehole-plug monitoring system.

The sensor selected for this project was a neutron/neutron sensor manufactured by Comprobe of Fort Worth, Texas. The selection was made primarily on the basis of price and power requirements. This logging tool consists of a 1-curie Am-Be source, a 4-atm ^{3}He detector, and pulse-amplification electronics, all contained in a stainless steel housing with a diameter of 1¼ inches and length of 50 inches.

Telemetry System

The Borehole-Plug-Monitor Underground Telemetry System is a wireless electromagnetic (EM) communication link that transmits information at relatively low frequency from significant depths in the lossy earth medium by using magnetic fields of generally vertical magnetic polarization between a pair of magnetic dipole (loop) antennas.

The general design goals considered were emplacement at a depth of 1500 m in earth with an average conductivity of 0.1 S/m for a worst-case operational system; and 750 m and 0.01 S/m, respectively, as a demonstration goal, to minimize prototype system size and cost. Specifically, a digital transmission system has been achieved that:

(1) Permits selective interrogation of multiple downhole systems which are in low-power standby until called either manually or automatically

(2) Can command the downhole system to perform a number of self-test or measurement-function operations

(3) Can remotely synchronize the downhole transmissions operation, which eliminates the need for complex highly stable circuitry downhole and permits fully synchronous detection at the surface

(4) Is basically transparent to the message and can be used with a variety of different sensors with an appropriate digital interface.

Operation

Figure 4 shows a block diagram of the data-transmission system. The downlink transmission is checked for bit and byte synchronization and parity and then decoded. If the error checks are satisfactory, a 16-bit word is serially transmitted to the instrumentation canister through the 12-conductor armored cable that connects the two canisters electrically. If data are expected from the canister, the wellhead receiver is switched onto the surface loop antenna shortly after the downlink transmission is complete. Data can be obtained from two sources in the borehole canisters. A telemetry test demands a retransmission to the surface with 4 bits of coded data that may indicate the health of the downhole system, or the command word may ask for retransmission of data from the instrumentation canister.

The sensor and data-processing system, shown in Figures 5 and 6, acquires data from the sensor and accumulates the pulses over an optimum period of time to get maximum resolution of data. The accumulated data are then presented to the transmission system, which transmits the information to the wellhead receiver. The system for acquiring the data is controlled in a systematic method by commands that are transmitted from the wellhead to the canister. These commands control when the sensor is turned on and off, reset the data accumulator, start the accumulation process, select the accumulation time and prescaling factors, and command the accumulated data to be transmitted to the wellhead. In addition to transmitting data the canister may be commanded to transmit a status word that shows some of the conditions that are being measured within the system.

The power efficiency of the transmission is enhanced by preprocessing the raw data from the sensor by a microprocessor system that operates in parallel with the data-acquisition system. The preprocessing of the raw data allows transmission of the significant data.

Tests

A successful integrated laboratory test was performed in February, 1979, using the prototype electronics. The tests were conducted following a Test Plan, and the results have been reported.[1] The test was monitored by personnel from Sandia, D'Appolonia, Develco, and IRT.

General Plans for FY 1980

The system will be calibrated and transported to the Nevada Test Site, where testing at the G tunnel in Area 12 will be conducted for the duration of the fiscal year. The underground system will be placed in the tunnel, and the wellhead electronics with the surface antenna will be located on the mesa above the tunnel.

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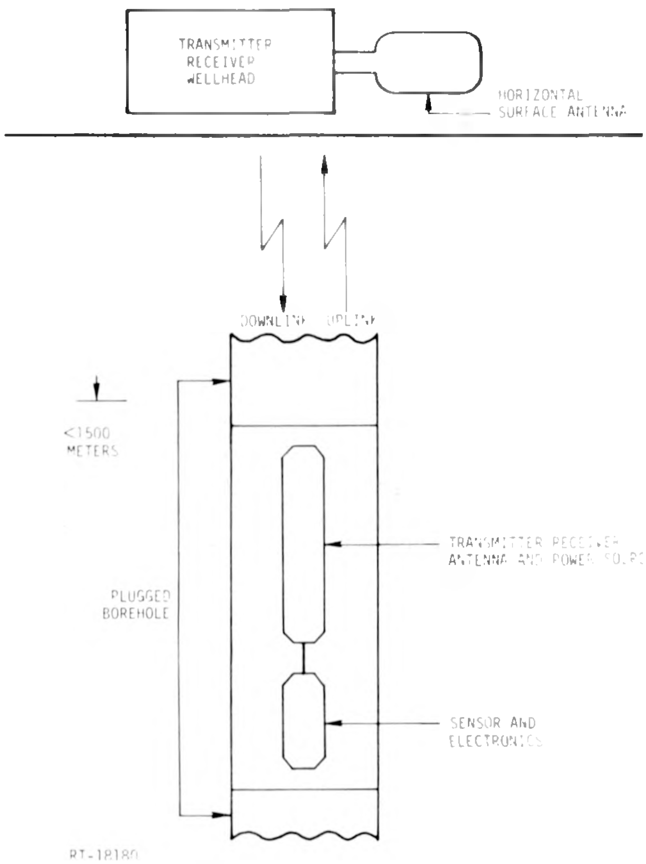

FIGURE 1. SYSTEM CONFIGURATION

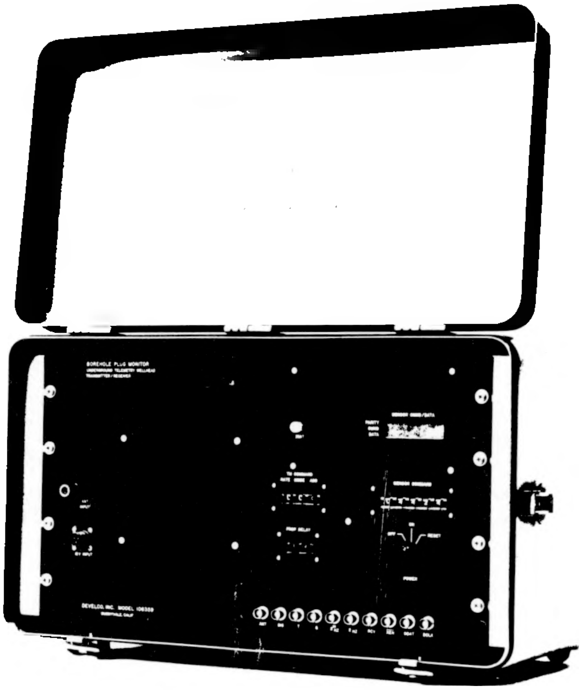

FIGURE 2. WELLHEAD ELECTRONICS

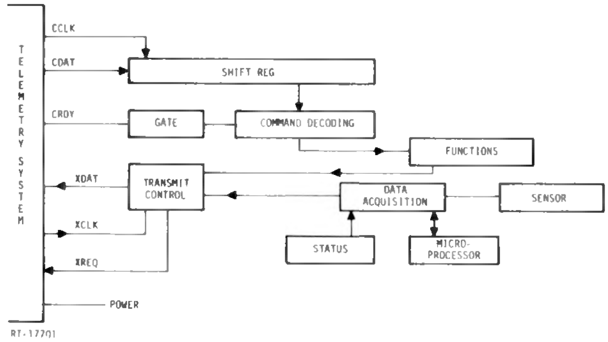

FIGURE 5. BLOCK DIAGRAM OF THE SENSOR AND DATA PROCESSING SYSTEM

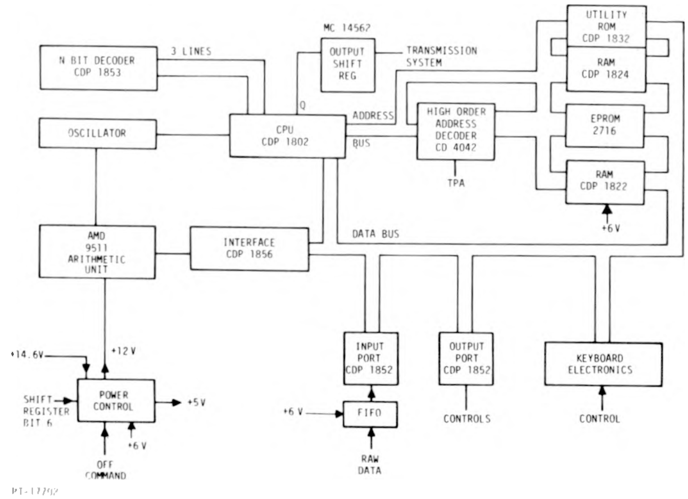

FIGURE 6. MICROPROCESSOR BLOCK DIAGRAM

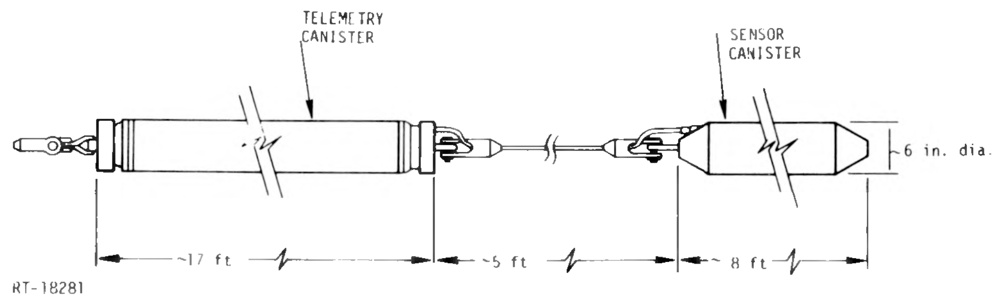

FIGURE 3. DOWNHOLD CANISTERS

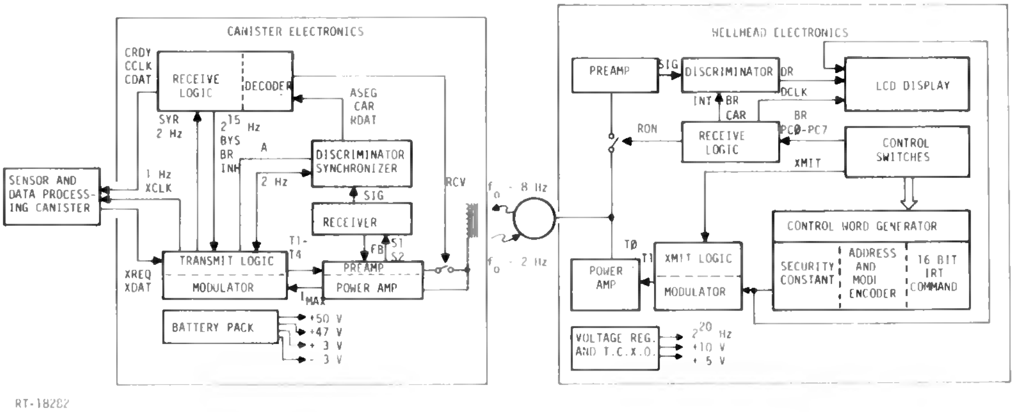

FIGURE 4. TELEMETRY SYSTEM

INSTRUMENTATION NEEDS FOR IN SITU TESTING PROGRAM

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The Ohio State University

Introduction

The objective of the in situ testing program is to provide data to assist with the development of theoretical models of geological repositories in order that long-term behavior of repository sites can be adequately predicted. The overall objective of this project is to determine instrumentation requirements for all current and planned experiments, and to recommend corresponding instrumentation research and development work, where needed. The scope of the project includes: (1) assessment of the environmental conditions and constraints on instrumentation, (2) identification of the instrumentation functional requirements, (3) assessment of current technology, and (4) recommendation of technology developmental programs.

This paper presents a summary of the in situ experiments and environmental parameters relevant to instrumentation, describes the instrumentation functional requirements, presents a summary of different instrumentation categories, and, finally, discusses the current project status and planned future work.

In Situ Experiments and Environmental Parameters

Several in situ experiments to obtain data needed to design and evaluate a long-term waste repository are either ongoing or planned for the near future. The geologic media under investigation in these studies cover a wide range. Currently, only heater tests are being performed, but spent-fuel tests are planned in the future. Table 1 summarizes the in situ experimental sites used as reference cases for this research program.[1-5]

The environmental parameters for each experimental site are in many cases different. In some cases the environmental parameters for particular sites are not currently available. A summary of the ranges of environmental parameters which may affect instrumentation is presented in Table 2.[2-14] A complete compilation of environmental parameters available from reports and publications has been prepared.[15]

Parameters Required by In Situ Experiments

The physical parameters of interest in the majority of the in situ experiments and the techniques commonly used for their measurement are summarized in this section.[2-5,16] In some cases the importance of the information is discussed. discussed.

The measurement of stress and stress fields may have the highest priority of all the mechanical properties. Stress is typically measured using hydrofracturing, flatjacks, vibrating-wire gage, or a thin diaphragm-type system. These techniques have the common problem of attachment to the structure in question. They also provide only single-point or local-type measurements. A method for measuring stress fields is currently unavailable.

The measurement of displacement to a resolution of 1 micron with no "stick-slip" problem is a high-priority requirement. Typically, displacement measurements are made by mechanical extensometers, although optical and laser interferometry techniques are being developed. However, the hostile environment encountered in a repository has inhibited the development of the more sophisticated techniques.

Because both mechanical and thermal properties are needed in many studies, the measurement of temperature is required. Temperature measurements are typically made using thermocouples and platinum resistance temperature detectors (RTD).

There is a significant need to measure fracture geometry in hard rock. Fracture geometry can be measured by displacement devices or acoustic transmission. These are local measurement techniques. The data obtained in hard rock are difficult to interpret because of the anisotropy of the medium. A satisfactory global-type fracture geometry measurement is not yet available.

Gas- and fluid-pressure measurements are also commonly required for in situ experiments. Pressure transducers typically used are piezometers or pressure cells.

There is a need to measure directly specific rock constants such as coefficient of thermal expansion, modulus of elasticity, heat capacity, etc. Currently, these constants are determined by measuring fundamental physical parameters and then calculating the constant.

In several experiments nuclear waste will be used. This will require measurement of the effects of radiation on rock parameters and instrumentation. Of highest priority is a noninvasive, time-dependent measurement of nuclide migration. This presents a difficult problem because of the very low rates of nuclide migration.

In addition to the parameters and phenomena discussed above, a number of experiments require measurements of permeability (time dependence is desirable), corrosion, moisture, humidity, deformation in anisotropic rock, heat flux, diffusivity, and pH. Often, this information can be derived indirectly from direct measurements of other parameters, such as temperature, displacement, and stress.

Research Plan

Based on the review of the in situ experimental data requirements it was concluded that many of the physical parameters and phenomena required for experimental analyses are common to the majority of the in situ experiments.[2–5, 16] It was also determined there are two broad classifications of measurements: (1) local or point measurements and (2) global measurements. The local measurements, as the name implies, refer to a measurement made by individual (or groups of) transducers at one particular location in the geological formation. If the formation is homogeneous, the measurements are characterized by good spatial resolution and are usable in theoretical models. However, for inhomogeneous geological formations, a local measurement provides data relevant to one particular region; due to the inhomogeneities, the measured phenomena or parameters may be significantly different in other locations. On the other hand, a global measurement often has poor spatial resolution but does provide a measurement of the average value of a phenomenon or parameter in a particular geological formation. For an inhomogeneous formation one must assume that, ultimately, measurements will be made using both local and global methods.

As a consequence of the generic nature of the required measurements and instruments, this research has been divided into five broad categories of instrumentation:

(1) Mechanical and thermomechanical. Rock-mechanics instrumentation as a specialty area plays a major role. In general these techniques fall into the classification of local or point measurements.

(2) Wave-propagation techniques, including telemetry. Wave propagation techniques include electromagnetic and acoustic waves and are usually classified as global type measurements.

(3) Nuclear techniques. Nuclear techniques include nuclide migration, measurement of radiation to analyze radiation effects, and measurement of radiation-related phenomena and mechanical properties.

(4) Data acquisition. This category includes any method used to acquire, manipulate, and store data. For example, the commonly used data logger will record data from a multitude of transducers simultaneously.

(5) Power supplies and batteries. There are unique requirements for supplying of power to the instrumentation in the remote and often harsh environment encountered in an in situ experiment.

Plans for Future Work

An assessment of environmental constraints on instrumentation and identification of instrumentation functional requirements is being completed. An assessment of current technology has been initiated and will be completed in the next 3 months. Development of recommendations for technology development will begin in the next 2 to 3 months and will be completed by January, 1980.

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TABLE 1. ONGOING AND PLANNED NWTS IN SITU EXPERIMENTS

Geologic Formation	Name	Approximate Time Period	Location	Geologic Type
Salt	WIPP	1979-1985	New Mexico	Bedded salt
	Avery Island	1977-1982	Louisiana	Domed salt
	STF	1982-1995	USA (TBD)	Salt
Rock	STRIPA	1977-1980	Sweden	Granite
	Climax at NTS	1978- ?	Nevada	Granite
	Eleana	1978-1979	Nevada	Shale
	Conasauge	1979-1979	Tennessee	Shale
	G-Tunnel at NTS	1979- ?	Nevada	Tuff
	BWIP	1979-1988	Washington	Basalt

TABLE 2. APPROXIMATE RANGE OF ENVIRONMENTAL PARAMETERS EXPECTED AT IN SITU EXPERIMENTAL SITES

Parameter	Estimated Range [a]	Units
Air temperature in mine (with seasonal variations)	(WIPP) 25 - NA	°C
Temperature (max) in vicinity of instrumentation	(WIPP) 43 - 400 (Avery Island)	°C
Room pressure	10^2 (WIPP)	kPa
Lithostatic pressure	1.5×10^4 (WIPP)	kPo
Humidity of air in mine	NA - 80 (Avery Island)	%
Water pH	6.5 - 7.5 (WIPP)	
Water content in repository rock	(WIPP) 0.019 - 4.0 (Eleana)	w/o
Corrosive agents in water (salt host rock):[b]		
Na^+	100 - 115,000	mg/liter
K^+	5 - 30,000	"
Mg^{++}	10 - 39,000 (Utah)	"
Ca^{++}	600 - 80,000 (Michigan)	"
Fe^{++}	1 - 2	"
Sr^{++}	5 - 2,000 (Michigan)	"
Li^+	NA - 1	"
Rb^+	1 - 1	"
Cs^+	NA - 1	"
Cl^-	200 - 250,000 (Michigan)	"
$SO_4^=$	1,750 - 3,500	"
$B(BO_3)$	10 - 1,200	"
HCO_3	10 - 700	"
NO_3	NA - 20	"
Br^-	400 - 3,100 (Utah)	"
I^-	NA - 10	"

(a) NA = not available.
(b) Values are for WIPP unless otherwise indicated.

REPOSITORY SEALING

OVERVIEW OF NWTS REPOSITORY SEALING PROGRAM

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Office of Nuclear Waste Isolation

When a repository for burial of nuclear waste has been constructed, and filled, it will be necessary to seal all the geological formations that have been disturbed. Further, the seals must function in a way that will prevent radionuclides from ever reaching the biosphere in harmful quantities. The technology for accomplishing this purpose must be explored in sufficient depth to ensure the safety and licensability of the repository. Effectiveness of that technology will be shown by mathematical modeling, laboratory analysis, and field tests.

Prior to selecting a site for the repository, an extensive geologic exploration program will be conducted. This study will ensure that the stratigraphy, structures, tectonic activity, hydrology, and other geologic conditions are suitable for safe storage of nuclear waste. The exploration effort itself will require some holes to be drilled into the repository area, and some preexisting holes may be present in the peripheral area. These holes will be sealed as soon as the selected site is approved.

Other holes will be needed during construction for access and ventilation, and tunnels will also be excavated. All of these penetrations must be filled and sealed where they go through formations with low permeabilities. Sealing of these penetrations will take place at the conclusion of the repository operating period.

A design for the plugs and seals will be developed in a conventional way except for one very important requirement. Some means must be found to ensure the plugs' effectiveness after 500 or more years of service.

Establishment of such a long design life is beyond conventional methods, so a geochemical program has been initiated to accomplish this purpose. Thermodynamic stability will be studied along with kinetic reactions of the sealing materials and the rocks they will contact. Studies of special geochemical phenomena believed to affect the plugs' longevity will also be carried out.

The design effort will receive information from geologic exploration on initial conditions within the repository area. Performance assessment and modeling studies will generate information on how those initial conditions (temperature, pressure) will change through the repository life. A computer modeling task will identify the consequences of any seal failure and thereby establish minimum acceptable performance for the seals. With this information in hand, the designers can specify materials and geometries for optimum performance. They will also be supported by an ongoing materials program in both laboratory and field testing.

In the laboratory, exhaustive testing and evaluations of candidate materials are being done. At the present time, most of the laboratory work is being done on cementitious mixtures. A wide range of additives is being examined.

Some field tests are under way at this time. Others are planned for the near future. These tests will attempt to secure information on performance of plugs in the field that is not otherwise available. They are investigative tests that will supplement and confirm the laboratory data. These field tests should not be confused with the prototype tests that will be done, in the field, later.

When final designs for the repository sealing system elements have been completed, plans call for construction of prototypes and field tests to establish their satisfactory performance. These tests will be done at sites that are candidates or potential candidates for repositories.

REPOSITORY SEALING DESIGN APPROACH — 1979

R. D. Ellison
D'Appolonia Consulting Engineers, Inc.

Introduction

Underground isolation of the radioactive wastes requires sealing of penetrations, such as shown in Figure 1:

- Exploratory boreholes
- Shafts and tunnels
- Underground rooms and chambers
- Horizontal exploration boreholes ahead of mining
- Nearby boreholes related to oil and gas, etc.

A recent draft report, "Repository Sealing Design Approach—1979" (under technical review), prepared by D'Appolonia, presents the first comprehensive design approach for repository seals. This initial qualitative approach is aimed at resolving issues raised by the technical community, licensing groups, and the public and is to be the "foundation" for all future studies leading to successful seals for specific repositories.

Figure 2 summarizes the major design elements. The starting points are (1) the design goals and licensing and quality-assurance requirements, and (2) specific site conditions. Acceptable materials and contruction procedures must then be sufficiently developed to show that the design goals can be met. A final step before actual design and installation is the verification of the suitability of the materials and procedures to actual site and design-criteria conditions by conducting laboratory and field tests and analyses.

Design Goals and Time-Dependent Considerations

Four alternative design goals considered vary from "returning the repository formation to its original permeability" to "providing seals which control releases to an acceptable level, through restriction of flow and adsorption of radionuclides in the seal zone." The seal zone is the sealed area and its immediate disturbed environment. The recommended qualitative goal is:

"The radionuclide migration rate through the seal zone is always less by a specified factor of safety than an acceptable level determined by a consequence analysis

$$R_t < \frac{R_{acceptable}}{\text{Factor of Safety}}$$

where: R_t = potential transport of radionuclides through the seal zone,

$R_{acceptable}$ = potential transport of radionuclides through the seal zone which is deemed acceptable."

This relationship to the consequences of radioactive release is similar to criteria being developed for determining suitability of entire repository sites. This design goal also recognizes the time-dependent factors associated with (1) long-term behavior of the waste form, (2) institutional controls, and (3) geologic and social events.

The factor of safety will depend on the adequacy of knowledge associated with:

- Material behavior
- Pressure, temperature, and environmental "loading" conditions
- Potential for geologic and social events
- Modifications of $R_{acceptable}$ due to institutional or site-analyses changes.

The factor of safety can be varied for different design periods based on determinations of how good the seal must be at any given time. Recommended design periods are:

- 0 to 300 years, when knowledge of material behavior is very high, transient loading conditions are reasonably predictable, and institutional controls can be available.

- 300 to 1,000 years, when the first major reduction in activity of the radionuclides occurs, confidence in material-property projections remains high, and, in some cases, engineered barriers are integral to performance; also, the probability of geologic events can be accurately predicted
- 1,000 to 100,000 years, when the activity of the radionuclides continues to drop slowly, but confidence in predictions of material behavior and geologic events is decreasing
- $>$100,000 years, when the activity of the radionuclides is decreasing toward innocuous levels and the scientific ability to predict material and geologic behavior is becoming low.

Characterization of Seal Conditions

An initial design consideration is characterization of:

- Host geologic environment
- Penetration
- Loading and environmental conditions.

The draft report provides basic characteristics of host formations consisting of bedded and dome salt, granite, basalt, shale, and tuff.

The characteristics of the penetrations themselves are very site-specific and include:

- Size, depth, direction, and age of the penetration
- Fluid characteristics and control devices
- Materials penetrated
- Physical and chemical properties of the rock
- Condition of walls and linings, if any
- Temperature gradient
- Extent of disturbed zone.

There are differing conditions for seals for boreholes, shafts, and tunnels. Major differences may include:

- Larger seals may have higher deformations and stresses.
- Boreholes may extend below the repository, while shaft depths are limited to the repository.
- Disturbed zone around large openings may be more extensive.
- Much more extensive behavior monitoring will have occurred in shafts and tunnels.
- Entire lengths of boreholes, but only segments of a shaft or tunnel, may be sealed by special materials.
- Rock properties and loads may be axisymmetric around a shaft, but very different around a tunnel.
- Gravity forces will tend to "squeeze" materials in a shaft, but cause separation at the roof of a tunnel.
- Future creep or load-induced deformations may cause larger stress-field conditions around a tunnel.

To maintain integrity for an extended period, each seal system must be capable of responding to all loading and/or environmental (static and dynamic) conditions which may occur due to site geology, the design of the repository, and/or the waste form. The general types are:

- Natural temperatures at depth
- Time-dependent rock deformations
- Geochemical and thermodynamic interactions of the rock, seal, and fluids, which may vary with geologic or climatic events
- Deformations due to earthquakes or tectonic activity
- Fluid pressure differences between aquifers
- Thermal loading from the waste form
- Radiation doses
- Gas pressures, if generated by the waste form.

Most of these conditions are time-dependent and can occur in various combinations. Also, additional loading conditions may be identified as repository sites are studied and as seal designs progress.

Material and Geometry Considerations

Seal materials and geometries must satisfy a broad range of sealing functions and locations. The functions are grouped into the following categories:

(1) Preventing the movement of permeants (water, oil, or gas) through the repository

(2) Preventing the movement of water toward the repository, particularly if the host rock is soluble

(3) Retarding the migration of radionuclides if permeant flow occurs

(4) Isolating water-bearing strata in order to maximize travel time of any escaping radionuclides.

Examples of important locations are shown in Figure 3. The importance of each function and location will change from site to site, but the material and geometric selection process can follow a general decision-making procedure.

Potential materials can be categorized according to primary functional properties, including:

- Flow-retardation (low permeability of the seal and its immediate environment)
- Radionuclide adsorption
- Deformation compatibility with surrounding rock
- Structural strength, which may be important in some cases
- Longevity.

Occasionally, one material may serve several or all of these functions. Frequently, however, a combination of materials may be required.

Some of these properties/materials have been investigated for the past several years. These programs have been expanded in FY 1979 to further document all pertinent properties within several years. Demonstration of longevity is of extreme importance to verify the adequacy of any design.

Figure 4 presents an example procedure for selecting the best materials for any sealing application. The matrix for a particular seal will include weighting and evaluation factors so that the various controls can be quantitatively compared.

Several basic shapes for seal components are illustrated in Figure 5. Some will be suitable with only certain materials or penetrations, while others may be appropriate to all conditions.

In many instances, it may be difficult to demonstrate by analysis and/or testing that any one single material or geometry will satisfy the stringent design requirements. Also, it can be very attractive to include redundant schemes within any given seal system to greatly increase reliability. Both of these factors provide a basic logic for considering the use of multiple-material/multiple-geometry seal systems. Figure 6 shows four examples.

Conceptual Design Approach

Following the selection of conceptual design alternatives, detailed analytical evaluations must be made to finalize an acceptable seal system. Major analytical considerations illustrated in Figure 7 include:

- Verification of seal integrity for all design phases for two conditions:

  Mechanical or structural "rupture" due to man's actions or geologic events

  Physical or chemical disintegration due to existing or potential future environment.

- Estimation of flow and radioactive release through the seal zone using two separate analyses:

  Initial evaluation of permeant flow

  Secondary evaluation of radionuclide migration

- Evaluation of the seal's ability to satisfy licensing requirements by conforming with the specified design goal.

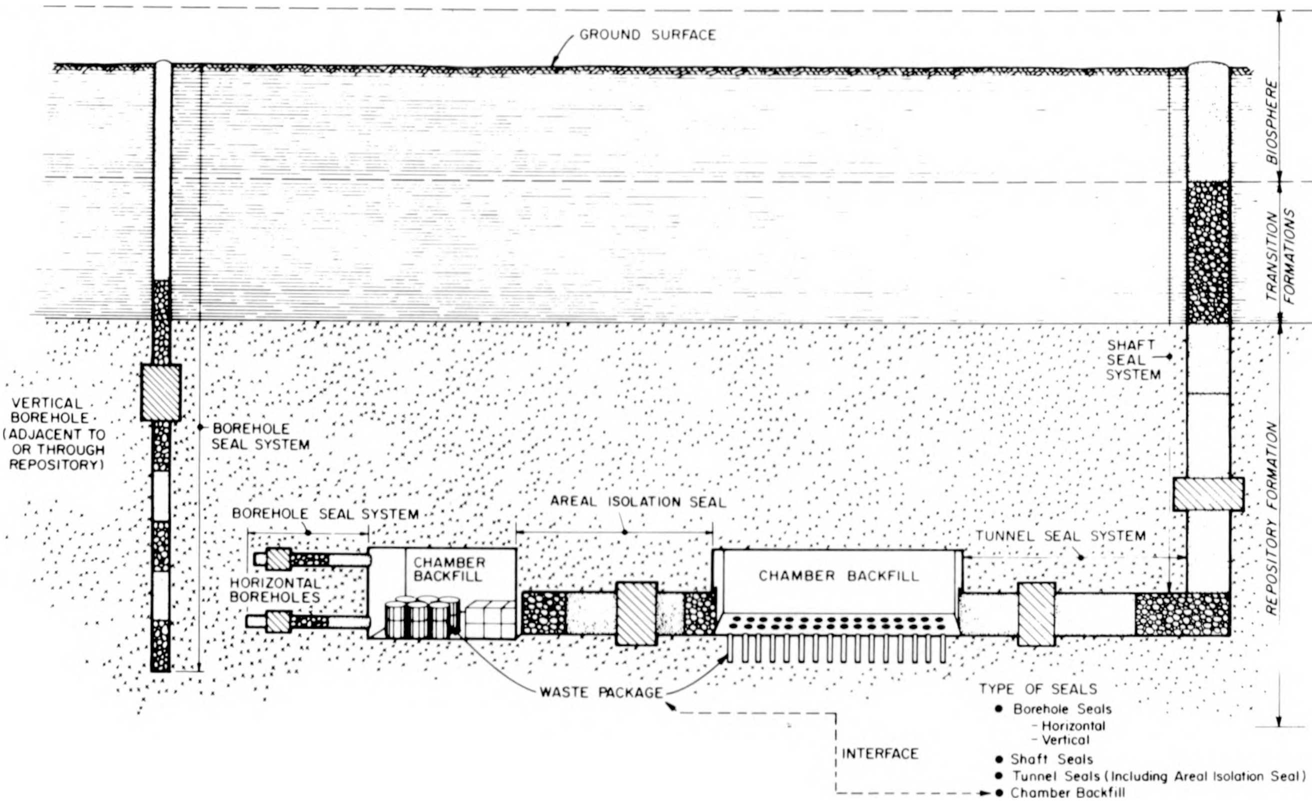


FIGURE 1. SCHEMATIC ILLUSTRATION OF VARIOUS SEALING APPLICATIONS

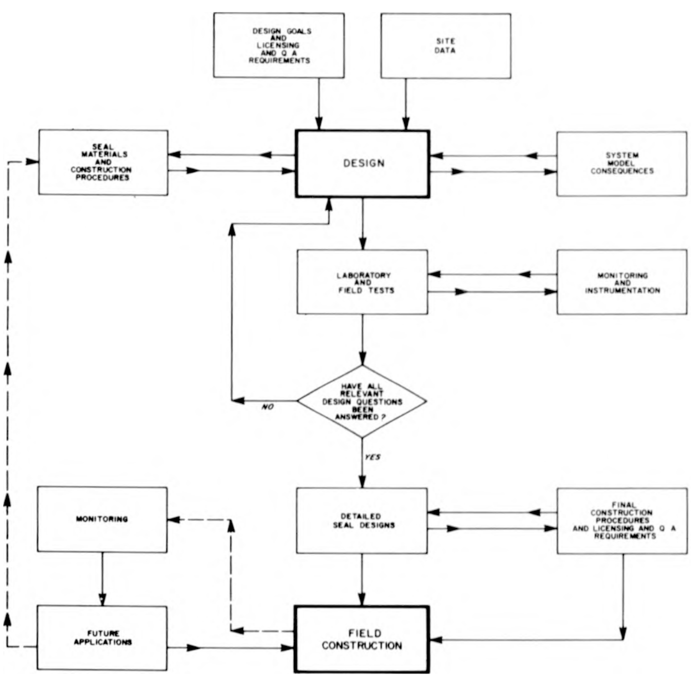


FIGURE 2. ELEMENTS OF DESIGN APPROACH

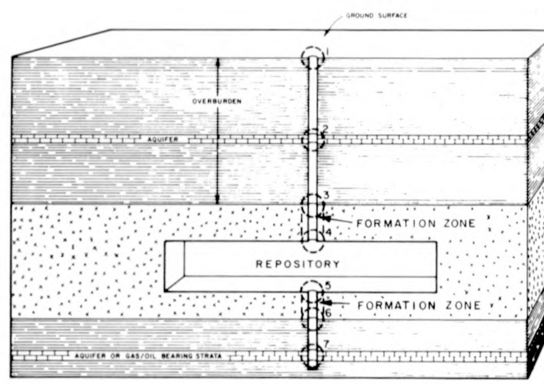


FIGURE 3. SCHEMATIC IMPORTANT SEAL LOCATIONS

MATERIAL TYPE	LONGEVITY	PERMEABILITY	SEAL / ROCK INTERFACE	FRACTURE REDUCTION IN DISTURBED ZONE	IMPACT OF PLACEMENT	ABSORPTION CAPACITY	DISPLACEMENT COMPATIBILITY AND/OR STRENGTH	PLACEMENT DIFFICULTY	QUALITY CONTROL	AVAILABILITY	PUBLIC ACCEPTANCE	COST	COMPOSITE EVALUATION
WEIGHTING FACTOR			(1)										
CEMENT GROUTS			(2)										(3)
GRANULAR SALT													
MOLTEN SALT													
COMPACTED CLAY													
CLAY SLURRY													
POLYMER CEMENT													
BITUMINOUS MATERIAL													
GLASS													
CERAMICS													
SYNTHETIC ROCK													

(1) WEIGHTING FACTOR
10 DESIGNATES HIGH POTENTIAL AS A MAJOR DESIGN AND EVALUATION FACTOR
1 DESIGNATES THAT THIS FACTOR MAY BE A MINOR CONSIDERATION

(2) EACH BLOCK DESIGNATION IS AS FOLLOWS

a	b
c	

a = DESIRABILITY FACTOR (DEGREE TO WHICH MATERIAL SATISFIES EVALUATION FACTOR)
i.e. −5 = VERY POORLY
+5 = VERY WELL
b = WEIGHTED DESIRABILITY FACTOR
i.e. DESIRABILITY FACTOR X WEIGHTING FACTOR
c = STATUS OF KNOWLEDGE
A INDICATES HIGH CONFIDENCE
E INDICATES POOR CONFIDENCE

(3) COMPOSITE EVALUATION CONSISTS OF THE FOLLOWING TWO NUMBERS

a	b

a = SUM OF WEIGHTED DESIRABILITY FACTORS
b = COMPOSITE INDICATION OF STATUS OF KNOWLEDGE

FIGURE 4. EXAMPLE MATERIAL EVALUATION MATRIX
Each matrix is appropriate for one penetration at a specific repository.

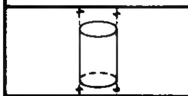
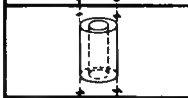
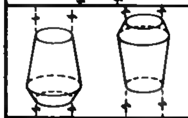
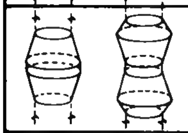
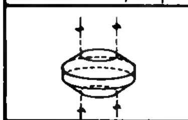
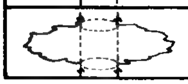
SEAL GEOMETRY	TYPE	EXAMPLE USES
	PRISMATIC (OFTEN CYLINDRICAL)	• FLOW BARRIER • ADSORBING COLUMN • INTERRUPT DISTURBED ZONE • FILLER • STRUCTURAL SUPPORT • DRAINAGE
	ANNULAR	• MONITORING POTENTIAL • USED IN TESTING • POSSIBLE CONDITION OF EXISTING LININGS • INTERFACE BONDS
	WEDGE KEY	• FLOW BARRIERS • STRUCTURAL SUPPORT • INTERRUPT DISTURBED ZONE
	DUAL WEDGE KEY	• IMPEDE FLOW FROM EITHER DIRECTION • STRUCTURAL SUPPORT • INTERRUPT DISTURBED ZONE
	CUTOFF COLLAR (FOR BOREHOLES) BULKHEAD (FOR SHAFTS)	• INTERCEPT DISTURBED ZONE • INCREASE POTENTIAL FLOW PATH • STRUCTURAL SUPPORT
	PRESSURE GROUT	• ISOLATION OF AQUIFERS (OR OTHER FLUID/GAS PRODUCING ZONES)

FIGURE 5. POTENTIAL SEAL GEOMETRIES

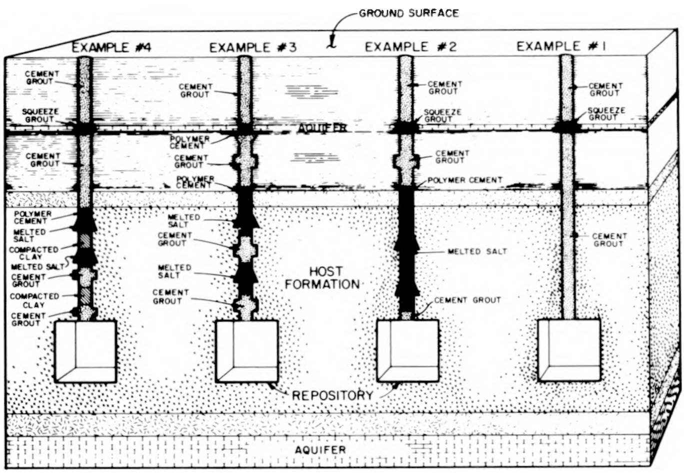


FIGURE 6. PRELIMINARY CONCEPTS OF MULTIPLE MATERIAL/GEOMETRY SEAL SYSTEMS

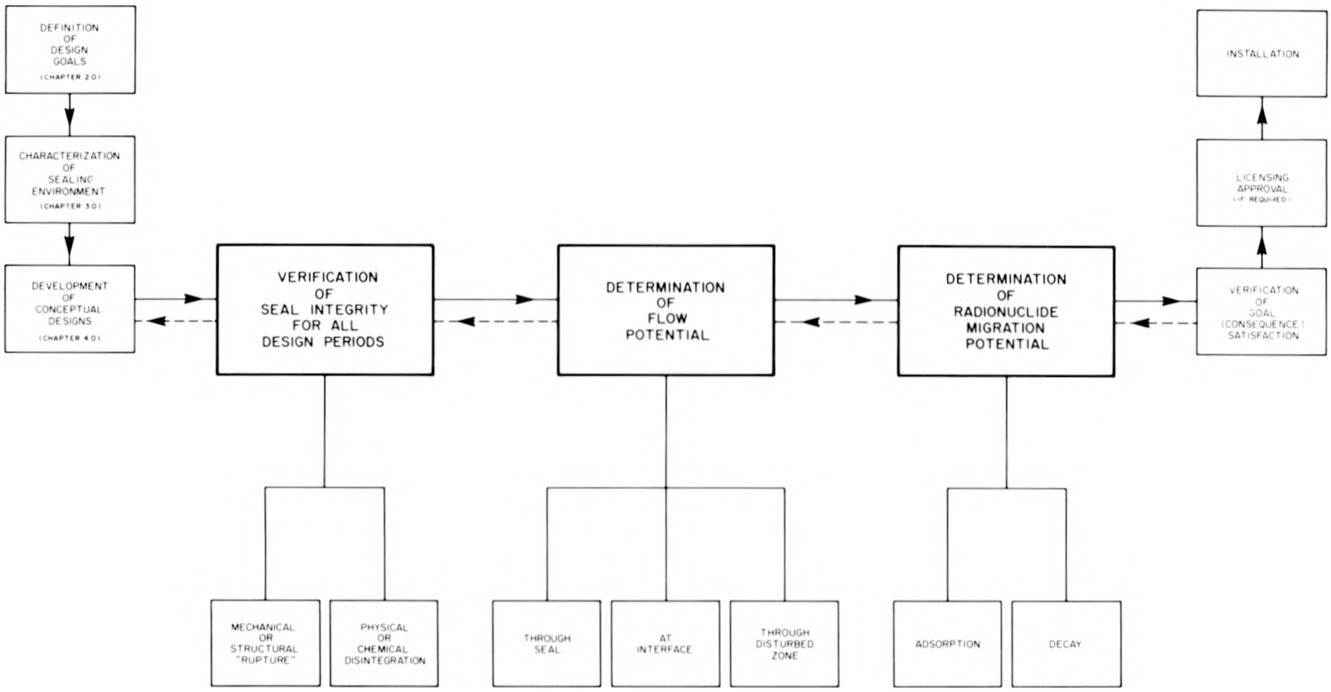


FIGURE 7. CONSIDERATIONS FOR DETAILED DESIGN OF SEALS

TESTING PROCEDURES AND INITIAL RESULTS FROM STUDIES ON THE EFFECT OF FLY ASH AND SALT IN MORTARS

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The Cement Technology for Borehole Plugging Program was established at Oak Ridge National Laboratory to assist in devising and testing cementitious mixtures suitable for plugging holes that may represent a hazard to a nuclear waste repository. Because of the extreme and exacting requirements of these plugs (they must form leak-proof bonds and maintain their integrity for time periods comparable to the surrounding rock formations), the ORNL studies have stressed the reliability of the physical and chemical measurements and the quality control that must be exercised on the ingredients used to make the plug material as well as the resulting solid. A major emphasis has been on establishing suitable methods, procedures, and techniques for measuring the physical and chemical properties of cementitious solids and geologic media. Procedures have been defined to determine the properties of the wet mix such as the rheology, bleeding characteristics, and set-times. The permeability, porosity, compressive-strength, thermal-conductivity, and shrinkage properties are determined for the solids after suitable curing.

The long-term objective of this program is to develop, from commercially available materials, cementitious mixtures that will eventually form thermodynamically stable solids. A systematic study of the effects of fly ash and salt on the physical properties of pozzolanic concretes and saltcretes was undertaken as the initial phase of this generic investigation. Pozzolanic concretes are known to have excellent durability and to be generally superior to ordinary Portland cement concretes. Saltcretes are under investigation because experience in the oil-well industry has shown that the presence of salt in cement grouts is required to ensure good bonding with sections of salt or shale encountered in the wall rock surrounding a hole.

Data have been developed on the effects of salt and fly ash on the physical properties of mortars in the wet and cured states. For example, the set-times of a standard mortar were doubled by the addition of 10 wt % salt. For saltcretes containing fly ash, the setting properties were dependent on the composition and concentration of the fly ash. Bleed characteristics were affected in a similar manner.

After curing under ambient conditions, the compressive strengths of mortars depended on the salt and fly ash concentration as well as the composition of the fly ash in question. Thermal conductivities generally increased with increasing density. Permeabilities of the mortars were very low, in the order of 10^{-5} to 10^{-7} darcy.

The most recent studies include work on the effect of temperature on the physical properties and phase compositions of mortars containing salt and/or fly ash, and the suitability of Sandia plug recipes proposed for the initial borehole-plugging field tests.

GEOCHEMICAL FACTORS IN BOREHOLE/SHAFT PLUG LONGEVITY

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Pennsylvania State University

Summary

Geochemical investigations have been initiated to address the factors that control longevity of plugging and sealing materials in a geochemical environment. The studies of borehole-plugging and shaft-sealing materials currently are investigating cement-based materials and their specific behavior in certain plug-rock environments. Factors controlling the extent of attainment of equilibrium of the plug components with time, and the rate of approach of the plug component chemical subsystem to a state of stable equilibrium with the total system are being investigated. The effect of these factors upon changes in physical, mechanical and thermal properties of the plug-rock system, and the consequent effectiveness of the plug in preventing radwaste material transport are the dominant features to be determined. Laboratory experiments, first preliminary, address the effects of anticipated pressure, temperature, and environmental (including specific rock type) ambients. Thermodynamic studies are used to project potentially stable reaction products under conditions similar to those projected for the repository sealing exits. Reaction kinetics and accelerated reactions are investigated for the purpose of determining the course of the reactions. Diffusion studies, detailed properties studies, and chemical and microphase characterization of the products of experiments are carried out. Characterization of old and ancient cements is performed to assess factors associated with longevity.

Introduction: Selection and Control Studies

The investigations under way are designed to answer a number of questions concerning the long-term performance of plugging and sealing systems. The information needed is categorized into different areas, including experimental, theoretical, and direct knowledge from the geosphere; the inter-relationships of the needed information are shown in the logic diagram of Figure 1. The scope of the work and its relationship to the information available from other sources have been described in a separate report[1], which deals with a broad range of factors considered important in assuring adequate performance and longevity of sealing and plugging materials.

The current status of availability and utility of relevant information[1] varies among the different areas, and was responsible for choices of initial studies. Table 1 outlines the current research in progress.

Materials selection and evaluation, much of it carried out in a separate but related study[2], result in choices of cementitious materials for the study of geochemical effects of longevity. Inasmuch as many of the final properties of solidified plugging and sealing materials are built-in by the initial mix characteristics during the materials selection process, thorough consideration is given to the mix parameters and placing characteristics, including compositions, mix characteristics and mixing procedures, viscosity, and other related factors. Hardening behavior is investigated to obtain quantifiable and reliable measurements, as is early-stage volume change at atmospheric and elevated pressures. Compressive strength, tensile strength, and microhardness are determined. Other physical- and thermal-properties measurements made include permeability, density, thermal conductivity, expansion, thermogravimetric analysis, and DTA.

Geochemical Investigations

Results from selection and control studies are incorporated and fed into the geochemical technical studies, as indicated in Table 1. One of the properties of considerable importance is the relative impermeability of plugging materials to gases and liquids.[3] An example of the permeabilities of some potential plugging materials under investigation is given in Table 2, where it is seen that most of the samples showed permeabilities that remained about the same or decreased with time, as would be desired. The properties of such materials will be followed for prolonged periods of time; accelerated studies are also in progress.

In one effort, a detailed study was carried out of a section of an 18-year-old 4-inch-diameter cement-based plug which had been placed to seal a borehole in the Salado Formation in New Mexico.[4] This was contrasted with some properties of current cementitious materials under investigation. The properties of the plug are summarized in Table 3.

Some currently prepared cementitious formulas incorporating sodium chloride which were under consideration for field tests in New Mexico were shown to have relatively superior properties to the older plug, as illustrated in the compressive strength data of Table 4.

The results of the study of the old plug, combined with recent laboratory studies, emphasize the necessity for preparing cementitious materials having optimal properties for use at a waste-isolation site. The borehole plugging material used in standard industrial procedures at the time the early plug was placed, although serving to maintain mechanical integrity in situ, seemed to fail many of the requirements for adequate sealing.[1]

Thermodynamic Studies

One approach to determining the thermodynamically stable end products of plugging materials in a geochemical environment is to determine certain thermodynamic parameters experimentally and then to calculate the free energies of formation; the lowest free energy for a particular composition then will be the most stable state. In the thermodynamic study, the first consideration has been the cementing phases in themselves, to be followed by more complex compositions. The required thermodynamic data for the major silicate cement compounds, dicalcium silicate and tricalcium silicate have been found to be inconsistent, and data will be obtained from the USGS, which has measurements in progress. Additional compounds, the calcium aluminates, are of importance as well. As shown in Table 5, there are inconsistencies in the data of previous workers and the current USGS data.[5] The data of the USGS are believed to be more correct. Additional work in progress is to carefully hand separate pure natural minerals, formulate models of potential reaction products, and also to synthesize pure phases which will be used for measurements. The precision of results of new measurements will depend highly upon the care of sample preparation and synthesis, and characterization of the stable mineral phases.

Ancient Cements

Ancient cements are presently being examined in order to determine long-term stability of man-made cementitious calcium silicate-based phases. Characterization of the cementing matrix phases and of the matrix-aggregate interfacial region is presently being carried out for five different materials from: (1) Cypress, ca 700 BC; (2) Roman (Carthage) ca 100 AD; (3) Casa Grande, Arizona, ca 1400 AD, (4) Chesapeake River canal, ca 1825-1850; and (5) a more recent one, 23 years old, from the Portland Cement Association. Examination by X-ray diffraction, optical studies of thin sections, SEM, STEM, and TEM studies, and DTA and TGA are among the methods used for characterization. A computer search was performed by Moore, et al. at ORNL to obtain information on archaeological or ancient cements, but failed to reveal much information in the literature that would be of use, beyond what was already known to the investigators. From these examinations of ancient cements insight into the possible changes taking place with longer periods of time may be gained, depending upon similarities and differences in the composition and ambients.

Plans for Future Work

The program for FY 1980 follows through from the currently initiated investigations, as outlined in the logic diagram of Figure 2. The major objectives are as follows:

To generate cementitious composite plugging and sealing materials that have high potential of approaching thermodynamic stability, or if not having full stability, then undergoing changes with time which would not be destructive; to follow this with studies of specific cement-rock combinations; to select and catalog essential thermodynamic data, and add the most important missing information; to perform control studies; to characterize starting materials, products, and ancient cements; to determine reaction rates and project changes; and to examine certain other special factors of probable importance for assuring the performance of borehole-plugging and shaft-sealing materials.

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(5) Robie, R. A., Hemingway, B. S. and Fisher, J. R., USGS Bull. 1452, 1978.

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TABLE 1. CURRENT STUDIES IN PROGRESS ON GEOCHEMICAL FACTORS IN BOREHOLE/SHAFT PLUG LONGEVITY

Materials Selection and Evaluation

I. Geochemistry/Longevity of Cementitious Materials (Borehole Plug/Shaft Seal)
   - A. Preliminary cement and cement-rock studies
   - B. Thermodynamic properties of cements; and cement-rock compositions; stabilities
   - C. Kinetics of reactions/accelerated studies
   - D. Properties and characteristics; including ancient cements
   - E. Support studies
   - F. Evaluative studies: longevity, shaft sealing

TABLE 2. SUMMARY OF EXTRAPOLATED LIQUID-PERMEABILITY AND WATER-PERMEABILITY VALUES FOR TYPE V AND CLASS C CEMENT SLURRIES CURED AT ROOM TEMPERATURE AND 60 C.

Sample	Permeability Darcy				
	14 Days	2 Months		6 Months	
	Gas	Gas	Water	Gas	Water
1A-V-0.73-1T-R	9.20×10^{-8}	7.60×10^{-6}	1×10^{-4}	Cracked	
2A-V-1-OT-Rt	2.41×10^{-6}	2.15×10^{-6}	—	8.00×10^{-8}	9.0×10^{-6}
2B-V-1-1T-Rt	4.60×10^{-6}	—	—	1.70×10^{-7}	1.5×10^{-7}
3A-C-1-0.73-T-Rt	—	8.55×10^{-6}	Cracked	6.00×10^{-6}	—
		1.60×10^{-5} at 81 psi	—	—	—
3A-C-1-0.73-1T-60	—	1.77×10^{-5} at 110 psi	6×10^{-7}	—	—
4A-C-1-1-1T-Rt	7.85×10^{-6}	—	Cracked	—	1.0×10^{-6}

TABLE 3. PHYSICAL AND MECHANICAL PROPERTIES OF BOREHOLE PLUG, AND PERMEABILITY VALUES FOR HOST ROCK AND INTERFACE

A. Bulk density		1.45 g/cm^3
B. Helium density		1.78 g/cm^3
C. Porosity		18.5%
D. Compressive strength		2.59 mPa (375 psi)
E. Solubility		~2 g/l in deionized water at room temperature
F. Elemental analysis of filtrate		
Cl	575 ppm	(gravimetrically)
Na_2O	310 ppm	
K_2O	216 ppm	
SiO_2	139 ppm	(by atomic absorption)
CaO	58 ppm	
Al	1 ppm	
Mg	1 ppm	
G. Elemental and XRD analysis of filtrate solids		

Elemental (SEM/EDX)	XRD
Cl	NaCl
Ca	KCl
Na	
Si	$CaCO_3$
S	Friedel's salt $[Ca_2Al(OH)_6]$ $[Cl \cdot 2H_2O]$
K	
Fe	C-S-H

H. Permeability	Salt	~2×10^{-6} Darcy
	Cement	~1×10^{-3} Darcy
	Interface	~3×10^{-3} Darcy

TABLE 4. COMPRESSIVE STRENGTH OF CEMENTITIOUS MATERIALS INCORPORATING SODIUM CHLORIDE COMPRESSIVE STRENGTH (M=5), mPa

Time, days	14		28	
27	50.1	± 1.6	64.4	± 3.7
60	64.4	± 1.9	67.1	± 3.2
90	47.0	± 4.0	61.0	± 2.8

TABLE 5. ENTHALPY OF FORMATION OF FOUR CALCIUM ALUMINATES FROM THE OXIDES CaO AND CORUNDUM, α-Al_2O_3 at 298.15K ENTHALPY, kj/MOL

Aluminates	Koehler, et al.[a]	USGS
C_3A	- 7.16 ± 1.67	- 6.6 ± 1.7
$C_{12}A_7$	-11.84 ± 1.84	-11.3 ± 1.8
CA	-15.96 ± 1.76	-15.4 · 1.8
CA_2	+ 6.36 ± 5.36	-37.3 ± 5.4

(a) Reference (6).

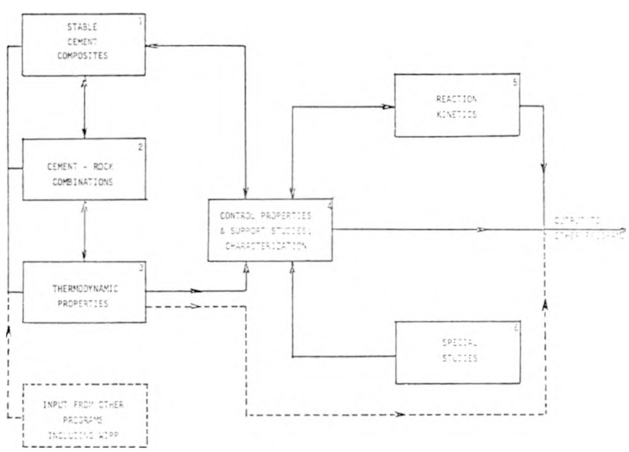

FIGURE 2. ORGANIZATION/INTERFACING OF TECHNICAL STUDIES (GEOCHEMICAL)

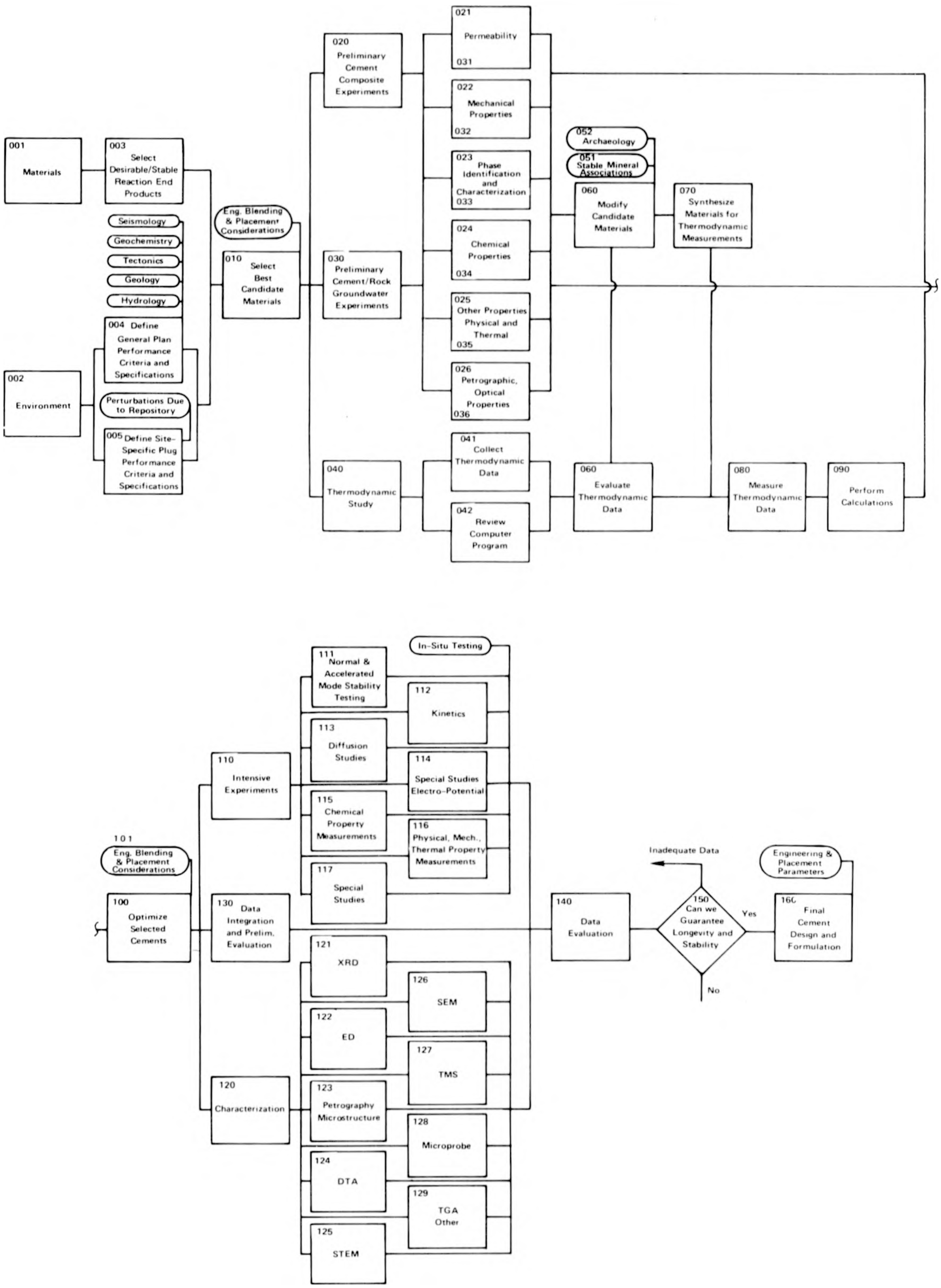

FIGURE 1. BOREHOLE PLUG LONGEVITY/STABILITY EVALUATION

FIELD TEST PROGRAMS OF BOREHOLE PLUGS IN SOUTHEASTERN NEW MEXICO

C.L.C. Christensen
Sandia Laboratories

This paper presents the status of the Borehole Plugging Program under way at Sandia, under the sponsorship of NWTS/ONWI, for the plugging of drill holes on and near the proposed Waste Isolation Pilot Plant (WIPP) in southeastern New Mexico (Figure 1). This program will provide the technology for plugging wellbores when required at repository decommissioning.

Current field-work emphasis is on the utilization of cementitious materials as the plugging material and the modification of existing oil-field emplacement techniques, and is supported by laboratory testing, consequence — analysis calculations, and an ONWI geochemical program. Another facet of the program is to provide demonstrable evidence of plug effectiveness based on the plug design models and field data. The present program is concentrated on small-diameter (6-12 inches) wellbores with intended expansion to large-diameter shafts as more experience is gained (Figure 2).

Although consequence-assessment calculations[1] being conducted at Sandia for the WIPP site indicate that even the most severe conditions of regional fluid flows through the repository via unplugged wellbores will not result in significant consequences to the public, unplugged wellbores in and near a repository must be plugged to develop the full assurance of the salt barrier. These same scenarios applied to other possible repository sites may not be as favorable, and hence a program to provide supporting evidence of the effectiveness of the plug and the development of the technology of plug emplacement and measurements for application to other repository zones must be pursued.

The proposed WIPP repository will be located in a bedded salt formation (Figure 3), located between low-yield waterbearing zones above and below.[2] The primary protection required of a wellbore plug is to minimize possible fluid migration through the repository. Consequently, the field effort is directed at understanding and quantifying the effect of a plug in reducing possible flows in the plugged region.

The first of the field efforts undertaken in FY 1979 assessed the effectiveness of an existing plug in media similar to those of the WIPP. A potash exploration hole, drilled and plugged in 1961, was intercepted at the ore horizon in a working potash mine. An approximately 0.8-m cubed section of ore, including the plug (Plug 217), was removed from the invert, sawed into nominal 20-cm-thick sections perpendicular to the plug axis, and one sample section each sent to the Army Corps of Engineers, Waterways Experiment Station, Vicksburg, Mississippi, and The Pennsylvania State University for analysis. The results of the petrographic analysis[3,4] indicate that the plug in this section did set up and was relatively competent with the regard to bond strength. However, the high water-to-cement (w/c) ratio (approximately 0.7) utilized during emplacement did not reflect the current technology of high-stength expansive grouts with a w/c near 0.3. This difference could account for the permeabilities of the plug/formation complex, which were on the order of 50 microdarcy, much higher than values of 10 microdarcy possible with more recently designed plug/salt complexes. Consequently, it was determined that even though the relatively long emplacement time of the plug should lead to valuable insight on plug performance, the unknowns associated with the original mix preclude definitive confidence in the 1961 technology. A need for additional testing was established and will be discussed in the final section under the Potash Core Test (PCT).

The second field effort in FY 1979 was the initiation of the Bell Canyon Test (BCT) in late March, 1979, with plug emplacement in August. In this effort, the objective was to evaluate a cement plug emplaced in a region with fluid pressure of approximately 2000 psi and to determine the leakage rate compared with that predicted from a knowledge of the plug and formation permeabilities (Figure 4). Given formation permeability on the order of 1 microdarcy (anhydrite) and plug permeabilities on the order of 0.1 microdarcy, the flow rates were predicted to be on the order of 80 cm^3/day through a 10-cm-radius plug. To observe these rates (approximately 0.2 cm/day at the plug top) in a realistic time frame, a minimum reasonable plug length of 150 cm (5 ft) was chosen for the test-plug design length. Although the original test plan called for an evacuated wellbore into which the plug would be emplaced on top of a timed deflatable packer, fluid influx from the upper zone of the wellbore required this plan to be modified.

The actual emplacement was done at a depth of approximately 1370 m (4500 ft) under a full wellbore brine head and permitted to set up for approximately 15 days prior to the release of the plug bottom packer. At this time, a tracer gas (SF_6) was released from below the plug via an included timer, and an initial determination of the plug integrity made. Once it was established that a gross failure of the plug had not occurred, the wellbore was cased to approximately 1330 m (4390 ft) and the brine exchanged for oil to permit detection of the lower zone aquifer fluids around the plug formation complex. The oil head is in the process of being reduced at selected intervals to verify the effect of the pressure differential across the plug on the observed flow volumes. The oil head will continue to be reduced until the full pressure differential is across the plug or until gross leakage occurs. The object of the test is to obtain data on plug effectiveness in a naturally occurring hostile emplacement.

Actual specifics of the test include:

Installation depth 1368 m (4490 ft)
Plug length 1.52 m (5 ft)
Plug-zone temperature 90 F
Plug-zone pressure 1897 psi

During the BCT fielding effort, the existing AEC-7 wellbore, drilled in 1974 to a total depth of 1195 m (3918 ft), was cored to a new total depth of 1438 m (4718 ft), intercepting the high-pressure zone in the Bell Canyon Formation to provide formation samples for laboratory analysis and to select an appropriate plug location. Prior to plug emplacement, extensive wellbore logging, upper salt-zone permeability measurements, and hydrologic testing were conducted to quantify the wellbore parameters as much as possible. This effort was undertaken to provide more information on the wellbore for future use in testing. Once satisfactory operation of the BCT plug is established, provisions for core-sampling of the plug are intended, and possible additional plugging tests will be conducted.

Future plans for the next year and successive years include continued testing of the BCT, removal of selected core sections from a three-year-old plug emplaced during the initial WIPP site-evaluation program in the Potash Core Test (PCT) previously referred to, and the establishment of a Surface Wellbore Test Bank (SWTB) for the emplacement and subsequent removal at nominal 2-, 5-, 10-, 15-, and 30-year increments of in situ cured plugs for long-term analysis. Other field tests that will address multicomponent plugs, multibarrier plugs, in situ diagnostic techniques in large-diameter plugs leading to shaft sealing experience, and in situ determination of formation parameters are now in the planning stage. A supporting materials and instrumentation program is also included in the Sandia Borehole Plugging Program.

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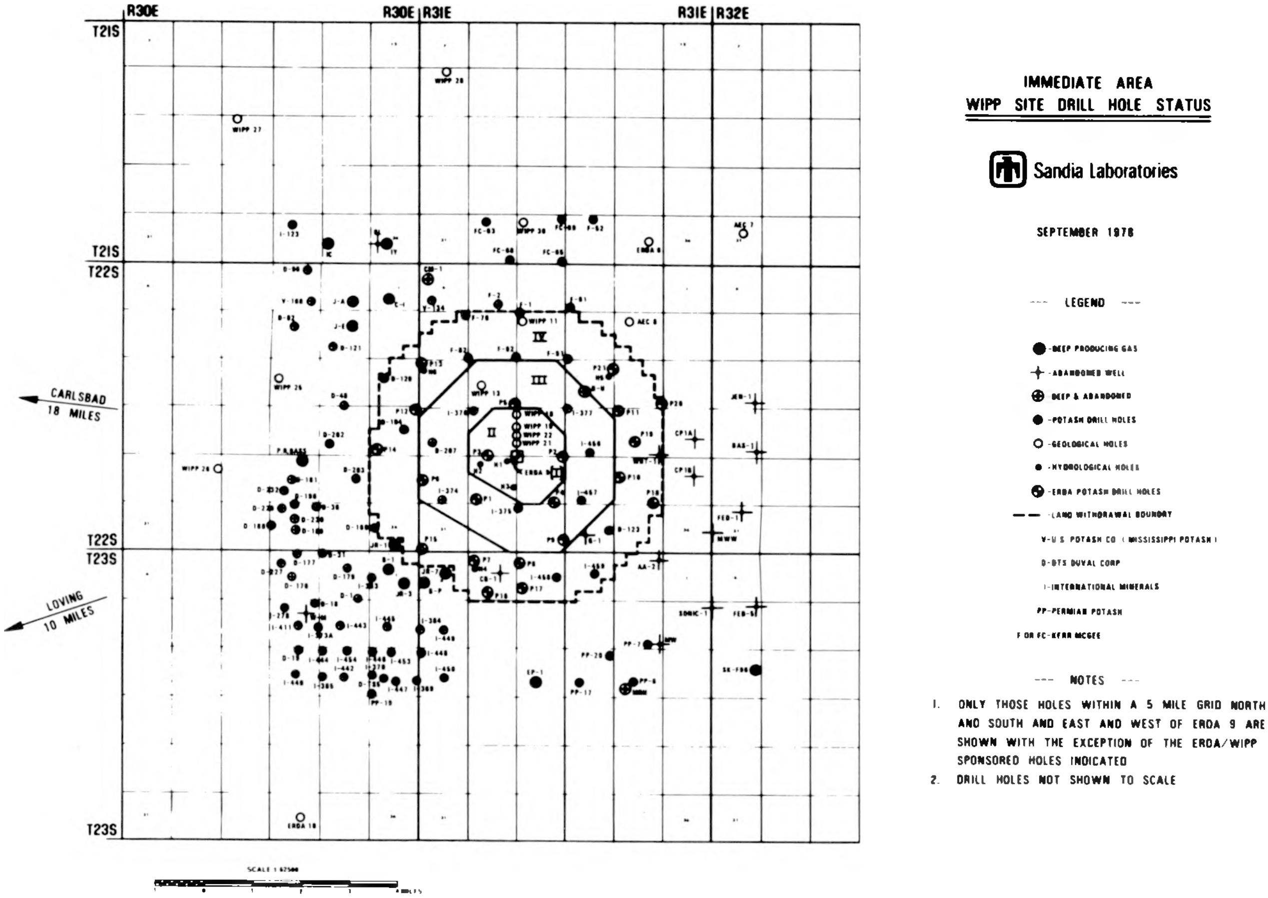

FIGURE 1. BOREHOLES NEAR PROPOSED WIPP SITE

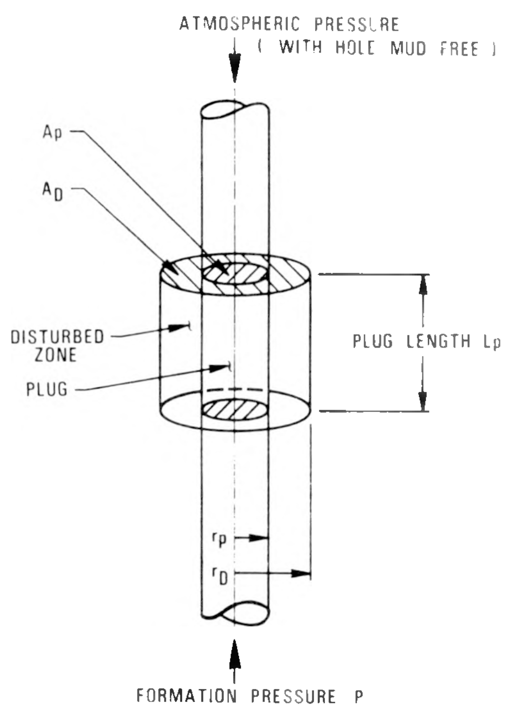

FIGURE 2. SCHEMATIC OF SMALL-DIAMETER BOREHOLE OF INTEREST IN CURRENT PLUGGING STUDIES

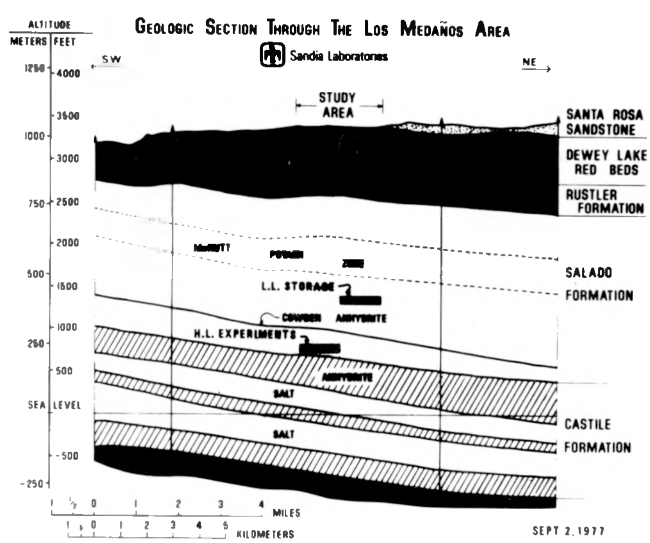

FIGURE 3. WIPP SITE BEDDED SALT FORMATION

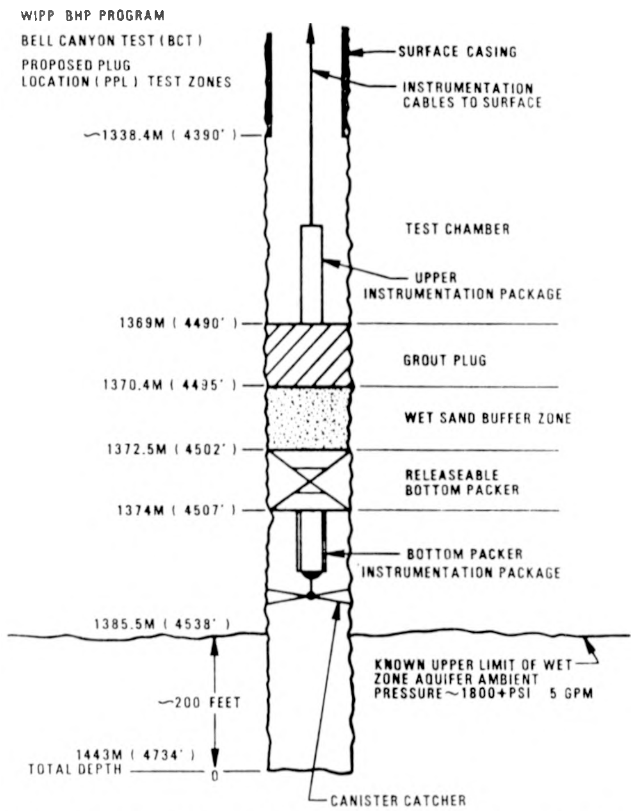

FIGURE 4. SCHEMATIC OF THE BELL CANYON TEST HOLE

ENCAPSULATION STUDIES

OVERVIEW OF ACTIVITIES AND DEVELOPMENT PLAN FOR REPOSITORY WASTE PACKAGES

J. A. Carr and S. J. Basham
Office of Nuclear Waste Isolation

The goal of the NWTS Program is to develop isolation technology that will provide a high degree of assurance that nuclear wastes can be isolated from the biosphere in a safe environmentally acceptable manner.

There has been a general belief in the scientific community that this isolation can be attained through placement in deep stable geologic formations.

Until recently, the waste package (defined as the waste and all associated man-made/emplaced components extending to the geology in the immediate vicinity of the waste) was considered (with the exception of waste-form stability and leachability) to play a minimal role in the isolation function beyond the repository operating period and the immediate period following repository closure. This "simple" or baseline package concept for spent fuel, for example, consisted of the conceptual components and performance functions shown in Table 1.

The succeeding five papers[1–5] in this session on Encapsulation Studies deal with activities and programs instituted or continued during FY 1979 which are responsive to this baseline package development effort. In addition, the EMAD support activities[1] and the development of a standardized experimental package for spent fuel[2] are supportive of NWTS in situ demonstrations and experiments being performed to develop the technology base to support geologic waste disposal.

During the past year there has been an increasing emphasis on a stronger performance role for the waste package in the geologic disposal system. The Interagency Review Group Report and initial ONWI interactions with NRC Regulatory and Licensing staff suggest that multiple engineered barriers should be utilized as backup to the containment features of the deep stable geologic setting and provide redundancy and/or enhancement of this isolation function. This emphasis increases the performance-function role of the package in the short term (i.e., emplacement and repository operating/closure period) with regard to accident scenarios and the initial repository qualification period and outlines a performance role for the package in the intermediate term. This intermediate term is loosely defined as the period of time between the end of repository operation/closure and the time when the shorter half-life highly radiotoxic and heat-generating nuclides such as ^{90}Sr and ^{137}Cs have decayed to nonhazardous levels. This time frame is approximately 600 years.

A number of ONWI studies[6-9] presented in these Proceedings are under way to better define the performance requirements of this multibarrier package. These studies will provide the criteria incentives and the operating-environment conditions from which a comprehensive waste-package design, development, and qualification program can proceed.

Conceptualizations of the various barriers contained in such a package revolve around the possible use of improved waste forms exhibiting improved environmental integrity and reduced degradation/leaching potential; canister fills/ stabilizers for nuclide immobilization, containment, criticality control, and deformation resistance; canisters of improved materials (metal or nonmetallic), possibly multilayered coatings for longer life and corrosion resistance; liner-canister and liner-geology annulus fills or pacs for inner-package barrier isolation from the environment, better waste containment, and improved waste immobilization.

At the present time, a detailed NWTS waste package development program plan is being formulated. The plan addresses both spent fuel and high-level waste and is intended as a definitive "stand-alone" document with respect to waste-package development for the waste-management program.

The package development program plan recognizes as front-end deliverables studies which lead to waste-package criteria, package performance incentives and functional requirements, and repository environment descriptions. The plan calls for (1) the development of conceptual-design approaches responsive to the criteria/incentives/ environment; (2) the development of functional performance description for the various "barrier" components of the package system; (3) a materials research and selection effort to identify candidate package component materials; (4) materials testing to determine performance under repository conditions; (5) package component tests; (6) interaction studies among package components, waste, and geology; (7) geochemical waste-rock interaction studies to determine multiparameter barrier effects; (8) package design and fabrication studies; (9) testing and qualification of package components and integrated barrier packages under normal, degraded, and accelerated conditions; and (10) organization of programmatic information in an output form to support repository licensing activities.

Additionally, the plan identifies programmatic interfaces among various DOE Headquarters program offices (ETW-P/ETW-I/ETW-F) and their respective lead contractors (ONWI/SRL/TTC-Sandia) and among NWTS programs (ONWI/WIPP-NTS/BWIP). Also interfaces are identified with other government agencies such as the USGS.

ONWI program activities for FY 1980 and future years will be structured to conform to this waste-package development plan.

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TABLE 1. BASELINE SPENT-FUEL PACKAGE CONCEPTUAL COMPONENTS AND PERFORMANCES FUNCTIONS

Component	Performance Function
Spent fuel pellets	Waste form from which eventual leaching and nuclide migration might proceed
Fuel-pin cladding	Barrier to fission-gas release, water and chemical attack on pellets, and geometry control for criticality
Canister	Operational handling, including retrievability and short-term containment of waste contents and short-term isolation of waste from external conditions
Air/helium canister filler	Helium for leak test of canister following packaging and minimal heat-transfer enhancement
Emplacement-hole liner	Canister corrosion protection barrier to moisture; potential lithostatic pressure protection; retrievability aid; nuclide-migration barrier in event of early canister failure
Liner-canister gap	Emplacement and retrievability aid
Liner-geology gap	Delay build up of lithostatic pressure on liner/ canister during operating period

EMAD SUPPORT FOR NWTS EXPERIMENTS AND DEMONSTRATIONS

R. J. Steffen, D. Ourrill, and J. B. Wright
Westinghouse Advanced Energy Systems Division

The EMAD facility, located in the southwest corner of the Nevada Test Site, has recently been converted, with minor modifications, into a facility used for the handling and encapsulation of spent fuel assemblies from commercial nuclear power plants. This facility was originally built for the assembly and post-operation disassembly and examination of nuclear rocket engines (hence the name: Engine, Maintenance, Assembly, and Disassembly). The versatile capabilities of the facility are available for support of NWTS/ONWI experiments and demonstrations.

The EMAD facility (Figure 1) contains a number of hot cells of varying size (Figure 2) each having the capability of handling highly radioactive objects from shielded work locations using a variety of remote handling devices. The main cell or Hot Bay, shown in Figure 3, is 43 meters long, 20 meters wide, and 22.5 meters high. Figure 3 shows the master-slave manipulators at the first floor viewing windows; the large, traveling wall-mounted handling devices; the overhead positioning system; the traveling bridge crane; and the railroad tracks running the length of the Hot Bay.

In FY 1978, modifications were made to the Hot Bay for the Spent-Fuel Handling and Packaging Program Demonstration. The objective of this Demonstration Program was to develop and test the capability to satisfactorily encapsulate typical spent fuel assemblies from commercial nuclear power plants and to establish the suitability of one or more surface and near-surface concepts for the interim dry storage of the encapsulated spent fuel assemblies. The modifications made to EMAD, together with associated equipment and components for the Demonstration Program, permit the receipt, visual inspection, lag storage, and encapsulation of Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) fuel assemblies, the leak testing of the completed canisters, and the emplacement of the canisters into interim storage configurations at EMAD. Many of the operations performed during this program are summarized in Figure 4.

The first PWR fuel assembly was received at EMAD and encapsulated in December 1978, and placed into storage in a sealed storage cask (SSC). Two more PWR fuel assemblies were encapsulated in January, 1979, and placed into instrumented drywells outside of EMAD (see Figure 5). The storage liner and canister configuration in the SSC is identical to that of the drywell with the exception that the configuration is above ground and surrounded by 3 feet of concrete for shielding.

Currently, work at EMAD includes the encapsulation of 13 PWR fuel assemblies for the Spent-Fuel Geologic Storage Test in the Climax Granite Stock at the Nevada Test Site. This work will continue into early FY 1980, after which several spent BWR fuel assemblies are scheduled for encapsulation and emplacement into interim storage at the EMAD complex.

The equipment and modifications made to EMAD to support the above programs are generic in the sense that they can be used (or easily modified) to fulfill encapsulation requirements for any experimental program involving the use of encapsulated spent fuel assemblies or any radioactive waste form. This is reflected in the fact that the design of a standardized canister specifically for use in NWTS/ONWI programs will be compatible with EMAD equipment and capabilities.

Presently, EMAD offers encapsulation and testing equipment, experience, baseline data, and qualified computer models which are available to support NWTS/ONWI programs. Examples of the existing features of EMAD include:

- A large receiving Cold Bay where equipment can be assembled, calibrated, and operationally tested prior to use in the Hot Bay.
- A machine shop, weld shop, and electrical laboratory to fabricate or repair equipment
- A hot-cell area consisting of a large Hot Bay, two smaller process cells, and 12 individual test cells (see Figure 2)
- Sets of master-slave manipulators at the shielding windows to perform remote operations requiring dexterous control

- Two mobile wall-mounted manipulators capable of lifting more than 272 kg when fully extended
- An overhead bridge crane with a rotatable main hook and an auxiliary hook having lifting capacities of 36 and 9 tonnes, respectively
- Ancillary handling equipment such as a floor-mounted turntable; portable turntables; and overhead positioning system; and special mobile, flat-bed train cars which can utilize the standard gage railroad tracks running the full length of the hot cell area
- Viewing periscopes at each shielding window capable of providing 10X magnification viewing and photographic recording
- A television network of remotely controlled portable cameras with video-tape recording capability
- Equipment to evacuate and backfill a canister assembly with any gaseous medium
- Equipment to perform a full-canister leak-tightness integrity check
- Equipment to verify that the outside surface of a canister is free of radioactive contamination
- An air-cooled vault for 24 canisterized fuel assemblies providing lag storage to enhance the sequencing of encapsulation operations
- A test apparatus in the West Process Cell to measure fuel-cladding temperatures and canister temperatures in an electrically simulated storage-cell environment
- An electrically heated drywell, instrumented with an extensive array of thermocouples, to measure the spatial temperature distributions on the canister and drywell surfaces and in the surrounding soil
- A series of in-line drywell storage cells (operating with encapsulated spent fuel assemblies) instrumented to determine temperature distributions on the canister, on the drywell liner, and in the surrounding soil
- Two fully instrumented, above-ground concrete Sealed Storage Casks (SSC's) which can accept encapsulated fuel assemblies
- A shielded transfer vehicle to transport and emplace encapsulated spent fuel assemblies in drywell storage cells.

To fulfill future needs of nuclear waste isolation programs, further additions to EMAD are planned for FY 1979 and FY 1980. Examples of these additions include:

- A fuel assembly calorimeter to accurately determine the thermal heat output of a spent fuel assembly
- A canister cutting tool to gain access to an encapsulated fuel assembly for post-demonstration examination
- Equipping one of the smaller test cells with instruments capable of performing physical measurements on fuel-assembly fuel rods before and after encapsulation and storage testing
- A decontamination station for remote removal of any radioactive contamination that might be present on canister external surfaces.

The capabilities of EMAD as a continuous packaging facility have been evaluated. The results of this evaluation show that with minor modifications to the existing facility (to enhance process flow and the sequencing of encapsulation operations) the EMAD facility can perform, at a minimum, 200 encapsulations per year. With some additional modifications, this throughput can be increased to over 500 encapsulations per year.

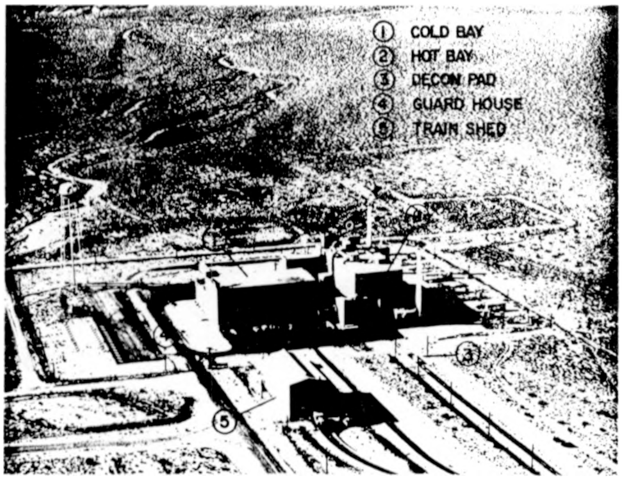

FIGURE 1. AERIAL PHOTO OF THE EMAD COMPLEX

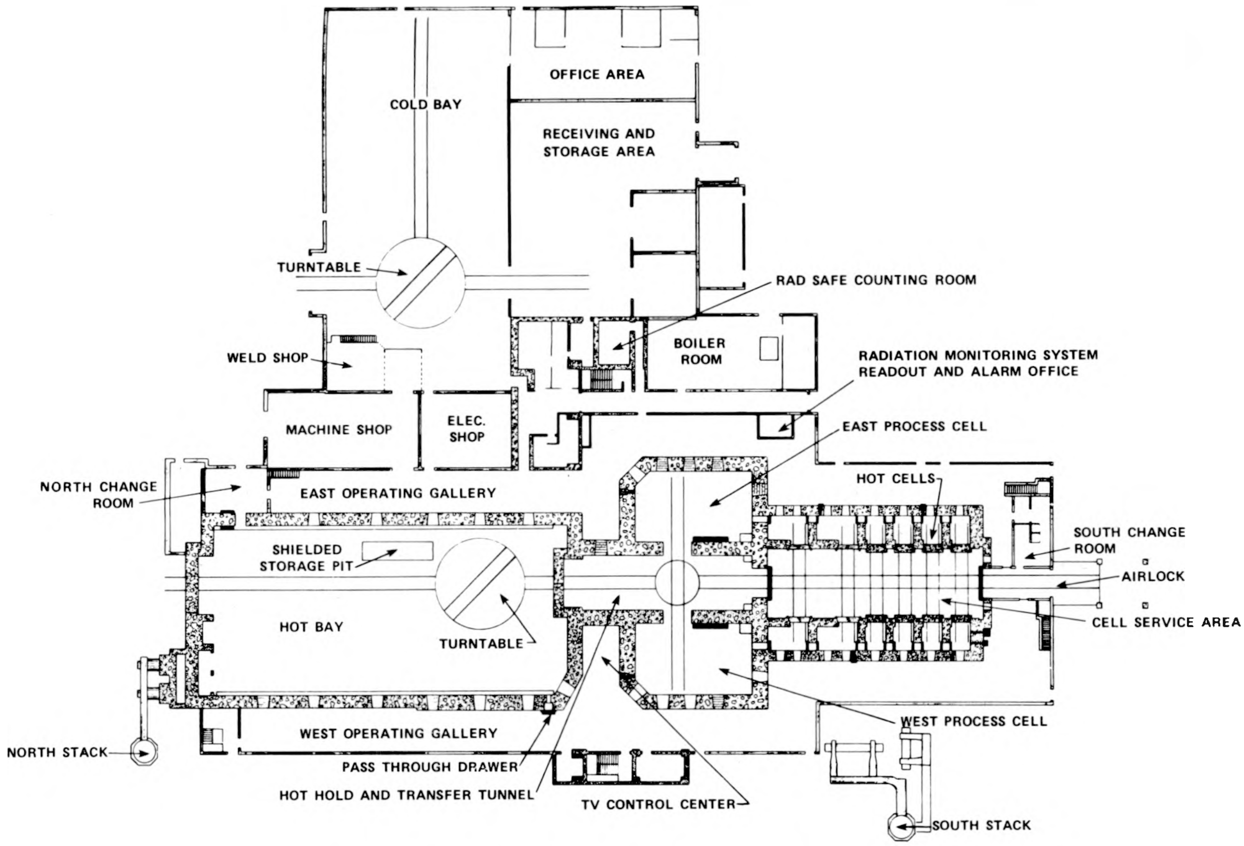

FIGURE 2. EMAD FACILITY MAIN FLOOR PLAN

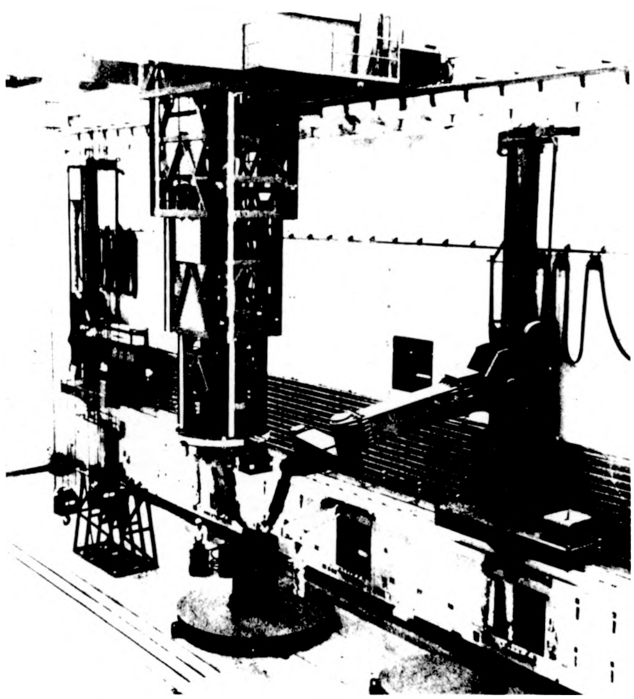

FIGURE 3. REMOTE HANDLING EQUIPMENT AT USE
INSIDE EMAD'S HOT BAY

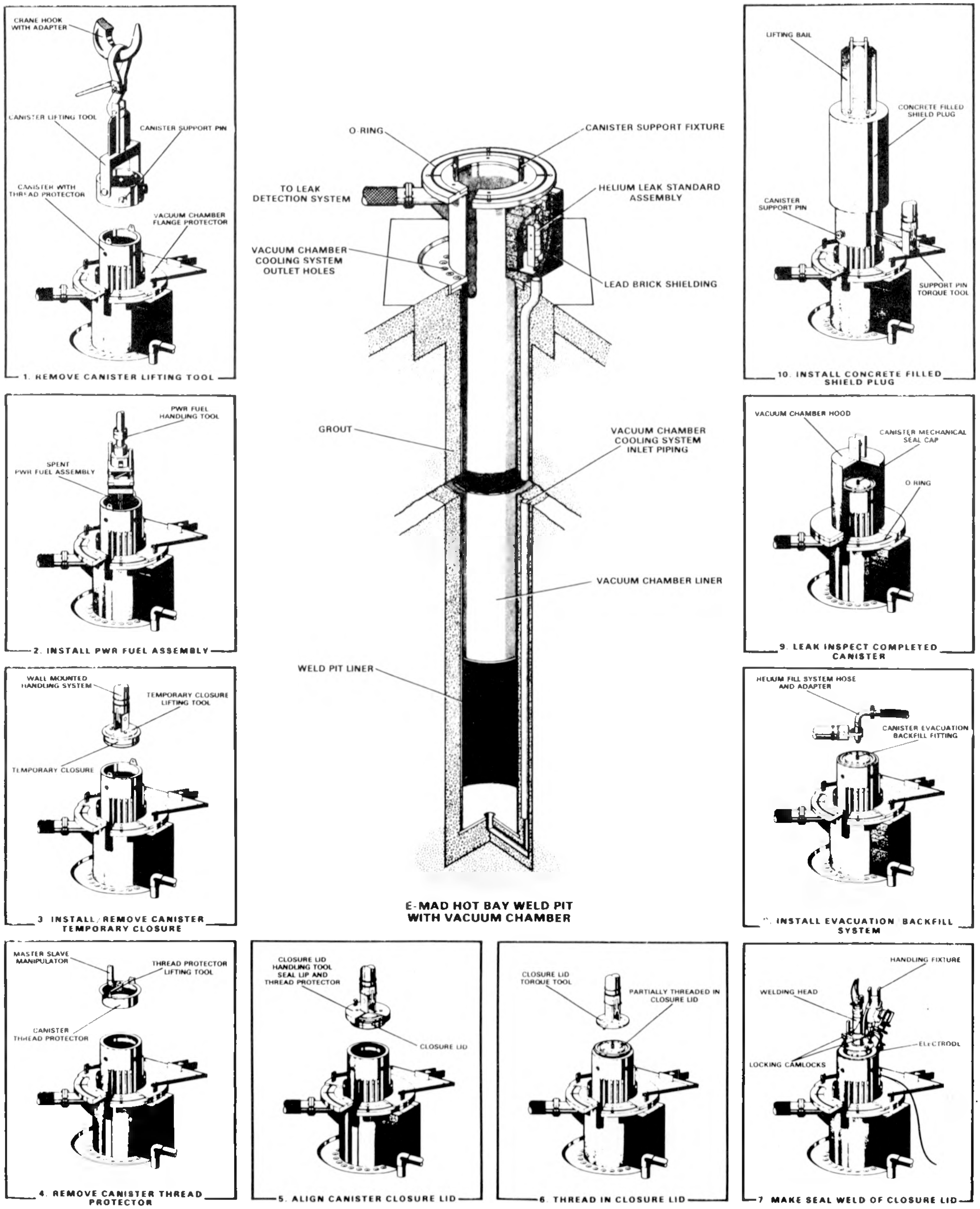

FIGURE 4. WELD PIT OPERATIONS SEQUENCE FOR CY 1978 SPENT FUEL HANDLING AND PACKAGING PROGRAM ENCAPSULATIONS

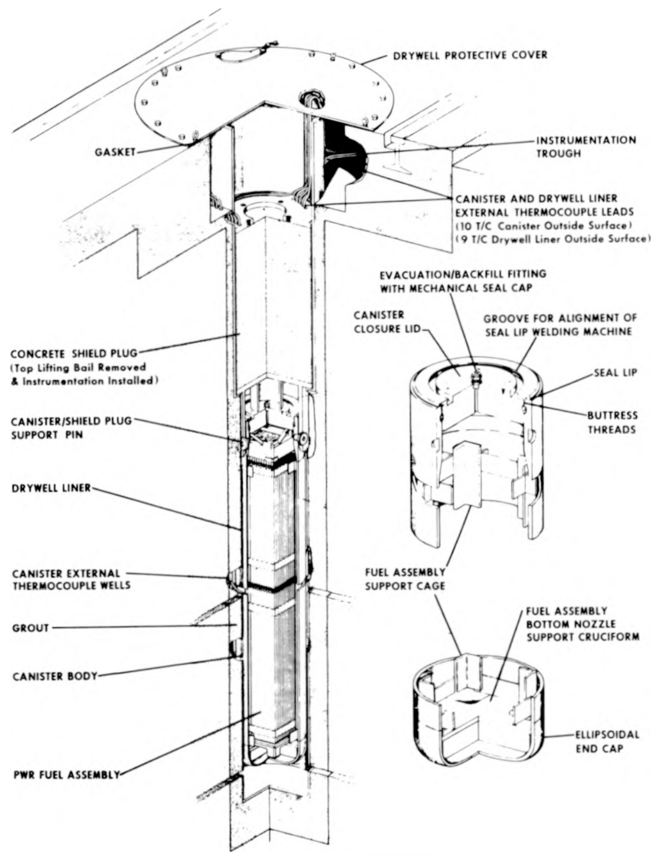

FIGURE 5. DRYWELL STORAGE ARRANGEMENT

STANDARDIZED EXPERIMENTAL PACKAGE FOR SPENT FUEL

J. M. Davis and R. E. Smith
Rockwell Hanford Operations

The Commercial Waste and Spent Fuel Packaging Program, being conducted by Rockwell Hanford Operations, has as its prime objective the development of waste packages for geologic disposal. In support of this objective is an activity to design a Standardized Experimental Demonstration Package for spent fuel, which can be used by any geologic demonstration with only a few and minor changes to suit site-specific conditions. This design effort will include analytical support, closure-weld development, and various tests to verify design and support the EMAD Safety Assessment Document modification. The activity will produce complete shop drawings and specifications to enable package fabrication whenever needed.

Containers fabricated in accordance with this standardized package design will be used by the Engine Maintenance and Disassembly (EMAD) facility at the Nevada Test Site to package spent-fuel assemblies for those contractors or agencies who plan to demonstrate spent-fuel emplacements in geologic media. The package will be sized to fit into presently available truck-mounted shipping casks. This standard design will thus reduce NWTS/ONWI costs for spent-fuel demonstration packages and enhance the analysis and comparison of test data accumulated by the demonstrations.

Activities During FY 1979

This effort was initiated during the first quarter of FY 1979 with the Westinghouse Advanced Energy Systems Division (W-AESD) in Pittsburgh, Pennsylvania, being selected to carry out the design program. W-AESD was selected because of experience in designing the packages used in the earlier SURF program demonstrations, which were assembled and installed by EMAD at the Nevada Test Site. W-AESD started work on the experimental package in May, 1979, and is scheduled to finish in May, 1980. (See Figure 1.)

Scope of Work

The scope of work for the Standardized Experimental Demonstration Package design contract is as follows:

> "Perform design, development, design verification testing, and produce specifications, drawings, material lists, and an EMAD Safety Assessment Document modification for the Standardized Experimental Demonstration Package for Spent Fuel in conformance with design requirements furnished by Rockwell. Coordinate and interface with Rockwell to assure proper tracking, scheduling, and interfacing. Hold appropriate design reviews and provide documentation."

A goal is to develop a design which is as close as feasible to the actual package which will be used for geologic disposal.

Design Requirements

Since performance, acceptance, and design criteria for geologic disposal packages are still being developed, this standardized demonstration package cannot be entirely prototypical of the final package design which will be selected for repositories. However, certain features, such as controlling codes and standards of construction, materials, closure-weld design, and, perhaps, lifting and handling provisions will conform to our present best assumptions as to final package design.

A complete explanation of the design requirements would be too lengthy for this abstract. Highlights are as follows:

- Performance. The package shall still provide containment following maximum credible accidents throughout a design life of 10 years.

- Codes and Standards. The package shall comply with the intent of the ASME B&PV Code, Section III, Division 1, Class 3 and be fabricated in compliance with 10CFR50 quality-assurance standards.

- Material. The package shall be made from low-carbon steel.

- Design. Requirements include the following:

  (1) The package must be transportable in existing, licensed, spent-fuel shipping casks.

  (2) A reference grappling tool and matching pintle will be chosen. The pintle can be changed to suit site-specific requirements, but early standardization is highly desirable.

  (3) The package shall be designed to contain one PWR or two BWR spent fuel assemblies.

  (4) The package shall be designed for a maximum wall temperature of 370 C.

  (5) The package design pressure shall be based upon release of the theoretical gas content from all fuel pins in a PWR assembly, and a maximum temperature of 370 C.

- Testing. Requirements include the following:

  (1) The package is to be drop tested for a vertical drop of 9.23 meters (30 feet), and an angle drop of 9.23 meters (30 feet).

  (2) The package shall be seismically tested at 0.7-g horizontal earth motion, in a configuration to simulate the lag storage pit at EMAD.

  (3) Except for test packages, no fabrication is included in the contract.

Conceptual Design

The conceptual design of the package has been completed. The welding system and closure weld-joint design have been chosen, and two preliminary 9.23-meter (30 feet) drop tests have been made to provide an initial confidence level in the choice of material as to size and type. The inside dimensions of available, licensed, shipping casks and the maximum dimensions of a PWR assembly dictate that the package must essentially use 29.48-cm (12-inch) schedule 20 pipe. The preliminary drop tests showed that this pipe material will be adequate, and that the package will not distort enough to preclude shipping it back to EMAD if it did indeed suffer such a drop.

The package will consist of a body made from low-carbon, mild steel, 29.48-cm (12-inch) schedule 20 pipe. End caps will be ellipsoidal and 0.84 cm (.375 inch) thick, with full-penetration butt joint welds. A handling pintle will be welded on the top end-closure cap, and an inner ring at the bottom end will support the spent fuel assembly.

The closure will be made by the plasma-arc method, as an autogenous butt weld. The weld will be certified to be in accordance with the intent of the code, and will be helium leak checked.

Work in FY 1980

The program is scheduled to be finished in May, 1980, and the major activities during those 8 months will include finishing the design and specifications, completing the weld-development program, completing the EMAD Safety Assessment Document modification for approval and all other supporting documentation, completing all EMAD interfaces, and concluding and reporting the seismic test. A Standardized Experimental Package for Spent Fuel that has undergone design verification will then be available for use in any geologic demonstration.

THE ROLE OF SPENT-FUEL CHARACTERIZATION IN THE DEVELOPMENT OF SAFE REPOSITORIES

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As a consequence of a Presidential directive in 1977 to defer indefinitely reprocessing of commercial light-water reactor (LWR) fuel rods[1], the Department of Energy (DOE) is considering LWR spent fuel as a reference waste form for geologic disposal. An understanding of the performance of this waste form under all anticipated disposal conditions is necessary to assure safe containment of radionuclides during the disposal cycle. The Hanford Engineering Development Laboratory (HEDL) has been contracted to develop a fundamental understanding of the behavior of spent-fuel assemblies and rods under all anticipated conditions imposed by transportation, packaging, and geologic disposal. A description of the coordinated program being implemented by HEDL to provide this understanding is described in another paper being presented at this meeting[2]. The subject of this paper is the role of spent-fuel characterization in the development of a fundamental understanding of spent-fuel behavior as a waste form.

Spent-fuel characterization is the process by which fuel assemblies and rods are examined to establish certain physical, mechanical, and chemical conditions of the spent fuel before and after testing. Pretest characterizations consist of nondestructive examination of test assemblies and rods plus destructive examinations of "companion" rods (defined later). Posttest characterizations include nondestructive and destructive examinations of test assemblies and rods. Information from these examinations will be used to identify and describe the behavior of the spent fuel for use in a computer code for spent-fuel performance prediction. This model will be developed from results of tests conducted on spent fuel in a variety of modes. Disposal demonstration tests will place unmodified spent-fuel assemblies in prototypic disposal arrays located in specific geologic media (e.g., granite, basalt, etc)[3,4]. Additional tests performed in the laboratory on individual fuel rods will investigate the spent-fuel behavior under simulated repository or design-limit conditions. Component tests performed on fuel and cladding as separate entities will add to the understanding of spent-fuel behavior necessary for the performance-model development. Detailed characterization of selected spent-fuel assemblies and rods before, in some cases during, and after disposal-demonstration field tests and laboratory tests will be an essential element of each test and critical to the success of that test. The characterization data will provide the primary basis or reference for interpretation and analysis of test results.

The method used to study spent-fuel behavior is the same as employed in other fields of scientific investigation, i.e., (1) establish the material starting condition, (2) perform the test (using known or measurable parameters), (3) establish the end-of-test material condition, and (4) compare the pretest and posttest conditions to determine the behavior. In this case, the materials referred to are spent-fuel rods where starting conditions will vary considerably due to differences in fuel-assembly fabrication and irradiation histories. End-of-test conditions will reflect a combination of starting conditions and specific parameters and conditions imposed during the tests. Pretest and posttest characterization examinations describing the physical, mechanical, and chemical conditions of the spent-fuel assemblies and rods are necessary to separate starting conditions from end conditions and to identify changes due to the test environment. Examinations are designed to provide both quantitative and qualitative information on the assemblies and selected fuel rods. Examinations performed on spent-fuel assemblies include a sip test, visual examination, and weight, length, flat-to-flat, and axial-bow measurements, as well as dosimetry (gamma and neutron flux). Nondestructive examinations on selected fuel rods include a visual examination; profilometry; gamma scanning; eddy-current testing; and rod weight, length, and bow measurements.

Detailed characterization of rod cladding and internal components requires destructive techniques that allow direct examination. For pretest rods, this is accomplished by selecting "companion" fuel rods from the same manufacturing and reactor loading batch with an in-reactor operating history as identical as possible to the fuel rods scheduled for disposal and laboratory testing. Nondestructive data gathered on the actual test fuel rods prior to the test are compared with correlated data (nondestructive versus destructive) obtained from companion rods to provide insight to the internal condition of the fuel rods. The fuel-rods destructive examinations include metallography of both the cladding and fuel, autoradiography, plenum gas-volume and content analyses, void-volume measurement, actinide assay, burnup determination, hydrogen analysis, microprobe examination, and mechanical property testing.

Each of the nondestructive and destructive examinations performed on the spent fuel contributes to an understanding of its character. Interim and posttest examinations allow probable identification of possible degradation mechanisms and quantification of degradation rates. It is only through an evaluation and comparison of this detailed pretest, interim, and posttest characterization information that geologic-disposal-test and laboratory-test results can be accurately interpreted. The tests will generate data over a wide range of conditions to facilitate the development of a comprehensive mathematical performance model. Interpretation and analysis of the disposal and laboratory test results in light of the fuel characterization are essential to the development and eventual validation of the spent-fuel performance prediction model. Thus, spent-fuel characterization plays a very important role in the development of a performance model that can be used in evaluating the ability of spent fuel to meet specific waste-acceptance criteria (e.g., the ability of cladding or fuel pellets to act as a line of defense for containment of radionuclides). Furthermore, this information provides a reference or basis for evaluating incentives for modification of the spent-fuel assemblies for long-term repository disposal purposes.

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Dup

SPENT-FUEL PREDICTION-MODEL DEVELOPMENT

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The extended time nature of nuclear waste disposal forces heavy reliance on long-term performance prediction to assure safe disposal of hazardous radionuclides. To one degree or another, this is true regardless of the waste form (i.e., high-level waste, low-level waste, or spent fuel). This paper is concerned with development of a performance prediction model for spent fuel. Technical studies are currently under way at the Hanford Engineering Development Laboratory to develop and validate an analytical systems performance model for predicting the behavior of commercial spent nuclear fuel from light-water reactors under all anticipated disposal-cycle conditions; e.g., transportation, handling, packaging, temporary storage, and geologic residence. Once developed, this performance model will be used to evaluate the suitability of spent fuel assemblies as nuclear waste forms and to identify and evaluate incentives for altering the waste form for geologic disposal in the absence of reprocessing. The technical studies highlighted here are concerned with the ability of spent fuel rods to contain radionuclides during the disposal cycle.

Typical environments expected during the disposal cycle range from inert gases to air of varying humidities to water containing dissolved gases and other chemicals. The chemical reactivities of species in these environments are affected by the radiation and heat (temperature) generated within the spent fuel from radioisotope decay. Superimposed on the chemical, radiation, and thermal environments are mechanical shock and vibration loads encountered during handling, transportation, and disposal (e.g., seismic loads) of the spent fuel.

A systems approach will be utilized to integrate the spent-fuel response to each environmental condition into a comprehensive performance model capable of predicting when the spent fuel form ceases to act as a barrier to radionuclide migration. The core of this performance model comprises calculations of spent-fuel temperatures and cladding stresses as functions of decay heat, initial rod internal pressure, and external boundary temperatures. Correlations will be developed to predict the response of the spent fuel to the pertinent environmental conditions. The significant spent-fuel responses will be added to the model core until a model compatible with the test results and theoretical mechanics is developed. Specific correlations to be developed and incorporated into the model include:

- Cladding stress-rupture correlations of time-to-rupture for predictions of initial loss of cladding integrity as a barrier to radionuclide release (i.e., cladding-breach formation) as a function of temperature, cladding stress, and cladding condition
- Cladding creep and strain-at-failure correlations for predictions of cladding strain and initial loss of cladding integrity as a function of time, temperature, cladding stress, and initial cladding condition
- Cladding inner and outer surface corrosion rates as functions of chemical and thermal environments to predict cladding wall thickness at a given time
- A cumulative damage correlation to account for time-varying spent-fuel temperatures, transient mechanical loads, and irradiation-induced material property changes
- Fuel oxidation rates with consequent effects on cladding stresses, release of volatile fission products, cladding-breach extension, and actinide redistribution.

Additional correlations will be developed and incorporated into the model as required, based on results from laboratory and field tests. Laboratory tests will investigate spent-fuel behavior under simulated normal and accelerated disposal conditions. These laboratory or diagnostic tests will include tests conducted on whole (intact) spent fuel rods to determine potential breach mechanisms and consequences; isolated component tests, performed on the fuel and cladding as separate components, will be used to study specific deterioration mechanisms and rates. Field tests or disposal demonstration tests will place the reference spent fuel form in prototypical disposal arrays located in specific geologic media. Characterization of spent-fuel performance during these tests will establish both the initial spent-fuel condition (parameters required for model input) and the reference performance (e.g., spent-fuel thermal response) as a function of in situ disposal conditions.

Presently identified rod diagnostic tests will be conducted on unmodified, prepressurized, and artificially defected spent fuel rods under controlled (laboratory) conditions to enhance reactions among fuel, cladding, and fission products. Within the scope of whole-rod testing, vibration and shock tests will be completed to assess spent-fuel fatigue and fracture integrity under dynamic loading conditions. Apart from revealing deteriorative mechanisms and the consequences of cladding failure, rod diagnostic testing is expected to provide the understanding required to combine specific analytical expressions into a comprehensive mathematical model capable of predicting spent-fuel performance during geologic disposal.

Isolated component tests, based on the logical separation of fuel from cladding, will establish rates for specific deteriorative mechanisms. Cladding performance studies currently identified as essential to performance-model development include biaxial stress-rupture, fatigue, fracture, and tensile material properties evaluations and corrosion tests. Fuel studies will be concerned with modes of deterioration, radionuclide release, and the long-term effect of fuel deterioration on breached-cladding configurations and actinide distribution.

The final area supporting the spent-fuel performance-model development is the disposal demonstration tests. These field tests will study the effects of radiation and thermal load on specific geologic host media (e.g., granite, basalt, salt) and provide the opportunity to investigate spent-fuel performance under in situ conditions. The variation in thermal properties of the different geologic media will result in unique temperature histories for all components of the disposal demonstration systems. The particular temperature history experienced will be a function of test-array geometry, thermal properties of the system (rock, concrete, external cooling, etc.), and decay heat of the spent fuel. The use of spent fuel assemblies with a range of initial decay-heat levels and the variation in thermal histories resulting from storage in different geologic media will provide a matrix of test parameters to complete development of a meaningful spent-fuel performance prediction model for actual disposal conditions.

SURVEY TEST OF CANISTER, GEOLOGY, AND FUEL CLADDING MATERIAL INTERACTIONS

Dup

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The Hanford Engineering Development Laboratory (HEDL) is conducting technical studies to develop and validate a performance model for predicting the behavior of commercial spent nuclear fuel in geologic disposal. An important aspect of the performance consists of the interactions among package, geology, and waste-form materials and the ability of the materials to coexist without loss of package or cladding integrity for extended time periods. A series of Materials Interaction Tests (MIT) is being conducted in conjunction with various spent-fuel disposal-demonstration programs. One test is in progress; two others are planned. The operating MIT currently resides within a spent fuel assembly in the Dry Surface Storage Demonstration (DSSD) test being performed at the Nevada Test Site (NTS). This paper provides a detailed summary of this first test and a status report on the MIT series.

The MIT in DSSD is a scoping test designed to provide generic materials-performance data for the spent-fuel prediction model and to demonstrate a testing technique utilizing representative disposal temperatures and radiation fields. Information relative to thermal and radiation effects on chemical stabilities and mechanical properties will be obtained from specimens representing potential canister materials, candidate geology samples, and fuel-rod cladding. Each type of specimen is included in three test groups: archive, thermal control, and spent-fuel emplacement. Archive specimens, retained for concurrent evaluation with emplacement specimens, will provide reference data needed to determine performance. Thermal-control specimens, placed within electric furnaces to reproduce emplacement temperatures, will provide information relative to thermally activated degradation and interactions. Spent-fuel emplacement specimens, held within a spent-fuel-assembly storage canister in DSSD at NTS, will combine both thermal and radiation effects. Comparison of the archive and posttest conditions of these groups of specimens will allow separation of thermal and radiation effects.

Thermal-control and spent-fuel emplacement specimens were fabricated, weighed and dimensionally measured prior to being loaded into capsules designed for installation into the control-rod guide tubes of a pressurized-water reactor (PWR) spent-fuel assembly. Generic geologic specimens include: basalt, granite, argillite, and welded tuff. Canister-material specimens were selected to represent classes of materials and include: austenitic stainless steels, low-carbon steels, nickel base alloys, and brass. Capsule interior atmospheres include: helium, air, and air plus water to represent a flooded repository condition. Individual MIT capsules vary in length from 1 to 3 feet, depending on the material property being studied. Top and bottom end caps welded to a stainless steel tube are designed for mechanical attachment to each other for ease of insertion and removal from the spent-fuel assembly. Encapsulated specimens are cylindrical with intimate contact between compatibility couples ensured by spring loading. Some capsules contain passive instrumentation to record peak temperatures and neutron fluence.

Following encapsulation of HEDL, one set of MIT capsules was shipped to Battelle's Columbus Laboratories (BCL) and inserted into thimble tubes of a spent-fuel assembly destined for DSSD emplacement at NTS. Companion thermal-control capsules were subsequently placed in a laboratory furnace at HEDL, where test temperatures are adjusted to track the spent-fuel test according to temperature data from the instrumented DSSD test. The DSSD test-specimen set was installed into a spent-fuel assembly in December, 1978, and the fuel assembly subsequently packaged and installed in the DSSD test array in January, 1979. Thermal-control tests were initiated in June, 1979. The tests will run between 2 and 5 years and attain a maximum temperature near 300 F. At the completion of the disposal tests, the MIT capsules will be removed from the spent-fuel assemblies and returned to HEDL for examination. When the companion thermal-control test capsules have accumulated an equivalent number of test hours, they will be removed from the test furnace for comparative specimen examinations. These data will be further compared with those from the examinations performed on archive specimens. In general, the posttest examinations are expected to include evaluation of intersurface compatibility using metallographic techniques; chemical and physical stability tests of geologic samples by weight measurement, dimensional and cover-gas analysis; specimen mechanical-property tests for modulus of rupture, fracture toughness, and compressive strength; and examination of rock and mineral samples for phase structure, grain size, and chemical or absorbed-water changes.

The second and third tests in the MIT series are being planned in conjunction with the Climax Spent-Fuel Test program and the Basalt Waste Isolation Program. These MITs will be designed to provide more specific data relative to disposal in granite, basalt, and salt.

SESSION IIIA
TECHNICAL STUDIES IN THE NWTS/ONWI SYSTEMS ANALYSIS PROGRAM

AN ASSESSMENT OF LWR SPENT-FUEL DISPOSAL OPTIONS

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Until 1977 the United States nuclear waste management program was focused primarily on a Light Water Reactor (LWR) fuel cycle involving reprocessing, recycle of uranium and plutonium, and ultimate disposal of nuclear wastes. The U.S. Department of Energy (DOE) is currently seeking alternatives to this program in response to a Presidential directive of April 7, 1977, to recommend alternative fuel-cycle systems which minimize the potential for unauthorized access to and use of nuclear materials.

This paper describes the results of a Spent-Fuel Disposal Study which compares alternate strategies for treating, encapsulating, and disposing of spent nuclear fuel, and contrasts them with a Reference Case which encompasses reprocessing and recycle of uranium and plutonium in LWRs. The study was conducted by Bechtel under a subcontract with NWTS/ONWI. It began in March, 1978, and was completed, with submittal of the draft final report[1], in July, 1979.

Study Definition and Bases

The three spent-fuel disposal systems (cases) defined and evaluated in this study are identified and contrasted with the Reference Case in Table 1. Case 1 represents the simplest process for disposing of spent fuel elements in a geologic repository. No processing beyond sealing the spent fuel elements in canisters is required. In Case 2, prior to encapsulation, the end fittings are removed from the spent fuel elements, and the fuel pins are chopped and voloxidized. This process permits the separation and collection of some of the volatile fission products and tritium contained in the fuel material, and allows some volume reduction in the encapsulated fuel material. In Case 3, the fuel material from Case 2 is dissolved in nitric acid, calcined, processed into vitrified form, and encapsulated in canisters for emplacement in the geologic repository. This case adds a further degree of immobility to the spent fuel and complete separation of the volatile fission products from the encapsulated material.

The scope of the spent-fuel disposal systems analyzed in the study is shown in the two fuel-cycle diagrams of Figures 1 and 2. The storage, transportation, processing, encapsulation, and disposal facilities and the transportation steps within the dotted lines are included in the study. All of these facilities and systems for each case were defined to the preconceptual design level to provide a basis for making the assessments and comparing the cases. Table 2 identifies a number of fundamental study bases employed to provide consistency in the study definitions and analyses.

Assessment Approach

To evaluate the study cases, a methodology was developed which produced either a qualitative or a quantitative measure of the merits of each case within each of the six assessment areas described below:

- The **technical feasibility** assessment produces a relative measure of confidence that the required spent-fuel disposal systems and facilities can be designed, constructed, and operated satisfactorily, based on the degree to which similar processes have already been developed and operated in practice.
- The **safeguards** assessment produces a relative measure of the expected risk to the public due to theft or sabotage of special nuclear materials.
- The **long-term criticality** assessment provides an analysis of expected public hazard due to hypothetical long-term criticality events occurring over geologic time in the repository.

- The **radiological impact** assessment produces a relative measure of the risk to public and operating personnel due to transportation, handling, processing, and storage of nuclear materials during facility operating lifetimes and over geologic time.
- The **retrievability** assessment produces a relative measure of confidence that all designated canisters can be satisfactorily retrieved from their emplaced position in the event the repository does not function as intended during an initial five-year demonstration period.
- The **economic** assessment develops the net levelized economic cost, in dollars per kilogram of heavy metal ($/KgHM), for each case on the basis of costs to society without allowance for profits, taxes, or insurance. Credits are assigned in the Reference Case for the value of recovered uranium and plutonium.

No attempt was made in the study to develop preference (or weighting) factors to sum the assessment of the individual areas for each case. Development of such preference factors is beyond the scope of this particular study.

Conclusions

While it was not the purpose of this study to combine the results of the individual assessment areas into a "bottom-line" summation for each case, it is helpful to view the results of all the assessments in one place — even in a summary form. Table 3 provides such a summary.

As a result of the information gained from establishing preconceptual designs for spent-fuel disposals systems, developing technical material for each study area, and making the assessments themselves, the following conclusions and observations are made:

- All of the spent-fuel disposal cases studied are technically feasible and there are no insurmountable technical problems inherent in their implementation.
- Of the spent-fuel disposal schemes, Case 1 is the simplest and it ranks well in almost all of the separate assessments.
- Transportation of all nuclear materials studied is feasible.
- There are no radiological or nuclear-safety impacts for the cases studied which should cause undue concern or which would subject the public to risks even approaching those which it routinely accepts.
- Long-term criticality is not a significant hazard and would not preclude the implementation of spent-fuel disposal schemes.
- The spent-fuel disposal cycle does not present an attractive target to a terrorist who seeks to fabricate a nuclear explosive device because the materials which might be stolen require considerable additional processing to be of value.
- The Reference Case is more attractive for diversion of nuclear materials because of the existence of separated plutonium. However, the risk of theft could be reduced considerably by coprocessing, spiking, or the co-location of a P/E facility with MOX fuel-fabrication facility.
- While the retrievability assessment in this study showed little differentiation among cases, the study shows that designing a repository with retrieval capability is feasible.
- The Reference Case is significantly more economical than the spent-fuel cases, but all cases are considered to be feasible at reasonable cost to society.

The Spent-Fuel Disposal Study described above was completed in July, 1979, and no further work on this particular activity has been identified for FY 1980.

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TABLE 1. ALTERNATIVE CASES FOR THE SPENT-FUEL DISPOSAL STUDY

	Alternative Strategy			
Operation	**Case 1**	**Case 2**	**Case 3**	**Reference Case**
Mechanical processing	None	End removal and chop	End removal and chop	End removal and chop
Chemical processing	None	Voloxidation	Voloxidation, dissolution	Voloxidation dissolution, solvent extraction
Recycle of U, Pu	No	No	No	Yes
Waste form encapsulated	Spent fuel	Voloxidated fuel and hulls	Vitrified HLW, including U, PU	Vitrified HLW

TABLE 2. SIGNIFICANT STUDY BASES

- Covers entire back-end of fuel cycle
- Hypothetical 5000 MTHM/yr system - all portions in equilibrium
- One-year storage in reactor spent-fuel basin
- Placed in repository at age 10 years
- Retrievability for 5 years after emplacement
- Geologic repository in a salt formation

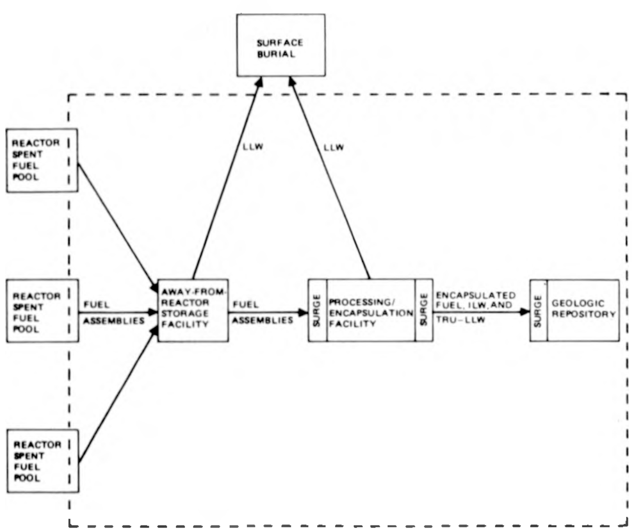

FIGURE 1. FUEL CYCLE DIAGRAM CASE 1, CASE 2, AND CASE 3

TABLE 3. ASSESSMENT SUMMARY COMPARISON OF RESULTS OF SPENT–FUEL DISPOSAL STUDY

Assessment Criterion	Rating Scale	Ref. Case	Alternative Strategy Case 1	Case 2	Case 3	Comments
Technical feasibility	1 to 5	1.9	1.7	1.9	2.1	Based on average of P/E facility and transportation
Safeguards:						
Sabotage	1 to 5	3	3	3	3	Based on transportation system as weakest link
Theft	1 to 5	3	2	2	2	Based on transportation and P/E facility as weakest links
Criticality	None	No ratings developed. Risk extremely low.				
Radiological impact:						
Near-term[a] occupational	man-rem /yr	3900	2500	3000	5100	Based on normal operation of 5000-MTHM/yr system
Near-term[b] nonoccupational	man-rem /yr	180	390	420	470	Based on normal operation of 5000-MTHM/yr system
Long-term	rem/event	340	90	100	44	Direct access drilling at 100 years; dose to maximum exposed drilling crew member
Retrievability	1 to 5	2.2	2.6	2.5	2.9	
Economics	$/kg	186[c] (154)	116	115	167	

(a) The average dose to individual workers ranges from 0.9 to 1.7 rem/yr depending on the case, the work facility, and the number of workers involved.

(b) For comparison the annual background dose is approximately 100 mrem/yr or 1.6×10^5 man-rem/year dose to the population living within a 50-mile radius of the site (based on 200 people per square mile).

(c) The reference case cost, before crediting with uranium and plutonium recovery values, is $186/kg. After crediting this becomes a net credit of $154/kg.

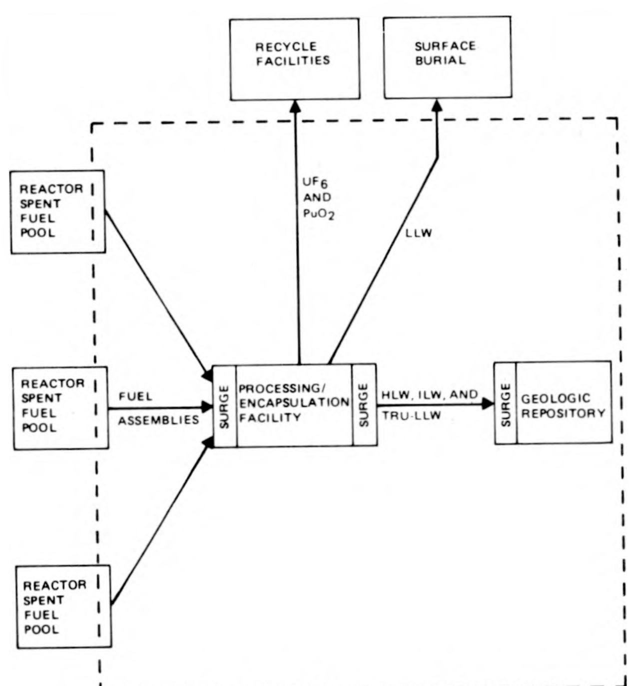

FIGURE 2. FUEL CYCLE DIAGRAM REFERENCE CASE

AN OVERVIEW OF NUCLEAR WASTE DISPOSAL IN SPACE

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The importance of national nuclear waste management dictates that studies of reasonable disposal alternatives be carried to the point that determinations can be made regarding their relative merit. Various concepts for the space disposal of nuclear waste have been studied by NASA.[1-18] Most recent studies by NASA's Marshall Space Flight Center have concluded that space disposal is feasible and that it can be carried out in a safe manner. The unique feature of the Space Option is the promise of total and permanent separation of certain wastes from the human environment. The Space Option is perceived not as a replacement for the mined repositories, but as an augmentation. Mined repositories, as well as all other options that leave radionuclides on or in the Earth's surface, must consider the long-term integrity of containment systems and potential release mechanisms, which are most difficult to quantify with any great certainty. Because of this, local public sentiment and perception of risk must also be considered for nuclear waste to be disposed of within any given local region. The Space Option could help alleviate some of these concerns by removing certain long-lived and difficult to confine radionuclides from the high-level wastes (HLW) and transporting them to a space destination. The residual HLW would still require disposal in a mined repository; however, the long-term hazard level of this residual waste may be significantly reduced, thereby improving confidence in the mined repositories. Therefore, the Space Option, in conjunction with mined repositories, may allow a more flexible and potentially lower risk waste-managment system.

The overall objective of the study of nuclear waste disposal in space is to assess the feasibility and determine the benefits of using a space option for augmenting mined repositories, which are the cornerstone of the national nuclear waste-management program. The current Space Option program activities are being supported and guided by NASA and DOE. Figure 1 identifies the government entities that are currently involved in managing and evaluating the results of the program effort. The newly formed (June, 1979) NASA/DOE Ad Hoc Coordinating Group has played an integral role in defining future program plans. The major study efforts that have occurred since 1972 are shown in Figure 2. Although not shown, studies were conducted by Battelle's Pacific Northwest Laboratories during the early 1970's. Research at Battelle's Columbus Laboratories has been supporting NASA in the space disposal study since 1977.

Over the years, many options for space disposal have been evaluated. Major considerations have included: (1) what waste should be disposed of; (2) what should the waste form be; (3) where should the launch site be located; (4) what space booster and upper stages should be used; (5) how can safety, together with viability, be maintained; and (6) where in space should the waste be disposed of? Figure 3 provides a summary of the options that have been and are being considered. The current reference space-disposal options are shown in the blocks; primary alternatives are indicated by an asterisk; and those options that are no longer considered viable have lines drawn through them. Consideration has been given only to technology that is likely to be available within the next 20 years. A pictorial overview of the current reference concept is given in Figure 4.

Disposal of all radioactive waste (e.g., entire spent fuel rods) in space is impractical with technology that is likely to be available within the next 20 years, because of the high launch rate required, the resulting environmental impact, high energy requirements, and high costs.[15] Thus, some form of waste separation will be required. The disposal of high-level in cermet form from a U-Pu recycle option to the solar-orbit disposal region appears to be reasonable, at least on the basis of the number of space flights that would be required. The Space Option could also be used to rid the earth of smaller quantities of specific individual radioactive wastes which are the greatest contributors to the long-term risks for geologic disposal.[11,15]

Several space disposal destinations have been considered in recent years. These include: solar system escape, injection into the sun, placement on the moon, and injection into a stable solar orbit. The first two options have been rejected since they demand more energy than necessary to assure permanent isolation from the Earth. Disposal on or in the lunar surface represents a very technically viable concept; however, the potential for adverse scientific and public opinion is high. The currently favored concept is injection into a circular 0.85 astronomical unit solar orbit, that is about halfway between the Earth and Venus. Orbital calculations indicate that at least for a million years, and probably more, this orbit is stable with respect to the Earth and Venus and would not intersect the orbits of either one.[13]

Several techniques could be used to place the nuclear waste into solar orbit. The concept receiving most attention (see Figure 4) would use the Uprated Space Shuttle (expected to be developed by NASA in the late 1980's) to boost the waste package, with its attached shielding, and an orbital-transfer vehicle plus a small second stage, to near-Earth orbit. The waste container would then be removed from its protective shielding and be propelled into the appropriate solar orbit. Along with the Space Shuttle hardware, the orbital-transfer vehicle and shielding would be recovered and returned to Earth for reuse.[17]

The FY 1979 program effort is summarized in Figure 5. NASA's Marshall Space Flight Center, Huntsville, Alabama, is performing the space-transportation and reentry-system design analysis. Most of the efforts of Battelle's Columbus Laboratories and Science Applications, Inc. (Schaumberg, Illinois) are directed toward providing support to NASA/MSFC in the areas of nuclear-waste payload characterization, the safety and health effects assessments, and long-term risk analysis. Various DOE laboratories have also supported the Battelle-Columbus effort in the area of waste-form definition. Battelle's Human Affairs Research Centers have also become involved by addressing social issues for ONWI. Bechtel is performing, for ONWI, a comparative assessment of alternative disposal concepts, including the Space Option.

Study activities being carried out during late FY 1979 and early FY 1980 have been preparing the foundation for the proposed 4-year program to develop and evaluate a concept for disposing of nuclear waste in space. Studies for the current year have addressed certain specific issues within each of the major work areas. Significant accomplishments to date (October, 1979) during this year's activity are as follows:

Waste Characterization

Recommended a cermet waste form having exceptional properties

Developed improved definition of the Hanford defense waste radionuclide inventory

Selected the Pacific Northwest Laboratories' defined PW-4b PUREX waste mix to be used for parametric studies of commercial waste payloads.

Systems

Assessed possible payload fabrication technologies and facilities requirements

Performed parametric analyses of space containment systems for the nuclear waste payloads

Assessed various space transportation-system options and recommended preferred concepts.

Hazards

Characterized on-or-near pad accident environments (blast, fire, and fragment impact) for the Uprated Space Shuttle and heavy-lift launch vehicles

Performed reference payload accident-response analyses (thermal and fragment impact) for on-or-near pad and reentry accidents

Performed preliminary assessments of deep-space rescue technologies

Accomplished preliminary assessment of dispersion of waste material (cermet and calcine) in space

Carried out preliminary assessment of long-term hazard reduction benefit due to space disposal

Integrated resuspension-effects model into computer code used to evaluate health effects from reentry accidents

Developed draft of system safety requirements for the space disposal mission.

Social

Completed updated assessment of social issues important to space disposal.

Economics

Performed cost analysis of various space transportation systems being considered.

Programmatic

Developed revised reference concept definition

Developed the Concept Program Plan.

Study activities which are soon to be completed are:

Hazards

An assessment of the health effects from certain reentry burnup accidents.

Programmatic

Preliminary identification of supporting research and technologies

Preliminary identification of licensing requirements

Preliminary identification of concept testing requirements.

An overall draft Concept Program Plan[18] for the Space Option has recently been developed to address the significant issues that were previously identified in the Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste. It covers intensive system and requirement definitions and impact assessments. The draft program plan addresses a four-year study activity (FY 1980 through FY 1983). Primary research functions during this period are:

Nuclear-waste systems definition and design

Space systems definition and design

Socioeconomic assessment

Regulatory affairs

Program management.

The first year (FY 1980) of the program activity will build upon the basis of previous study results, with emphasis on identifying and assessing all reasonable options within the space-disposal alternative. At the end of the second year of study (FY 1981), preferred space-disposal options will have been selected as candidates for further consideration in the study. The FY 1982 effort will emphasize preliminary system designs of the unique space and nuclear-waste systems and facilities and result in the selection of a baseline disposal concept. The fourth year effort (FY 1983) will consist of further refinement of the baseline concept. On the basis of the integrated results of the program study activity and other pertinent factors, it will be possible for DOE and NASA to recommend continuing activities (if any) to the administration.

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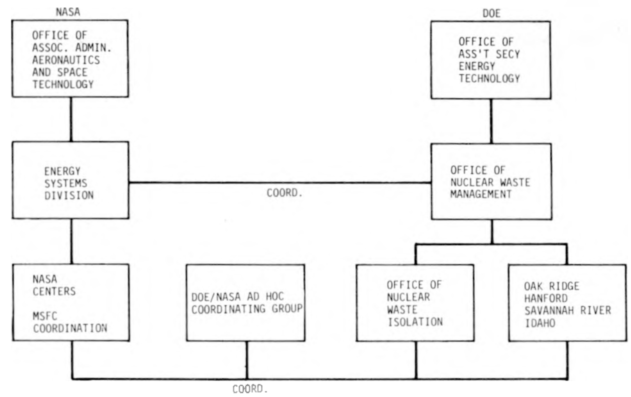

FIGURE 1. NASA AND DOE ORGANIZATIONAL RELATIONSHIPS FOR THE PROGRAM ON NUCLEAR WASTE DISPOSAL IN SPACE

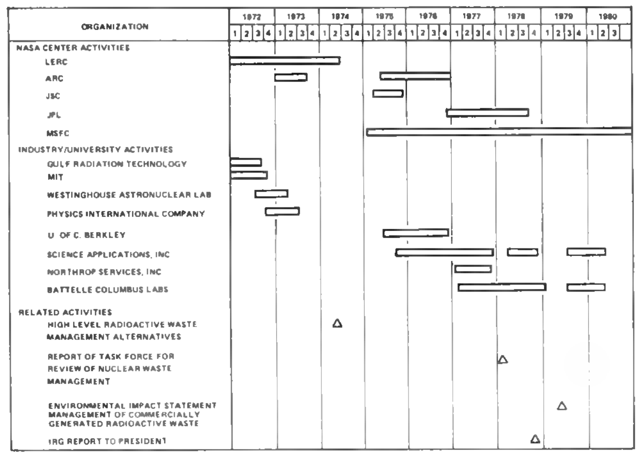

FIGURE 2. SCHEDULE OVERVIEW FOR THE PROGRAM NUCLEAR WASTE DISPOSAL IN SPACE

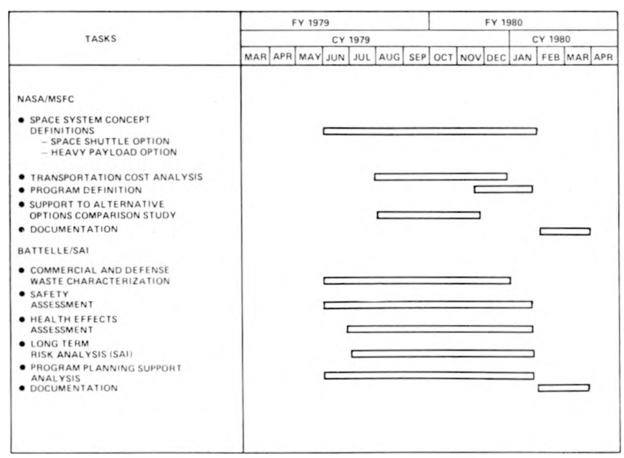

FIGURE 5. FY-1979/1980 STUDY ACTIVITY FOR SPACE-DISPOSAL OPTION

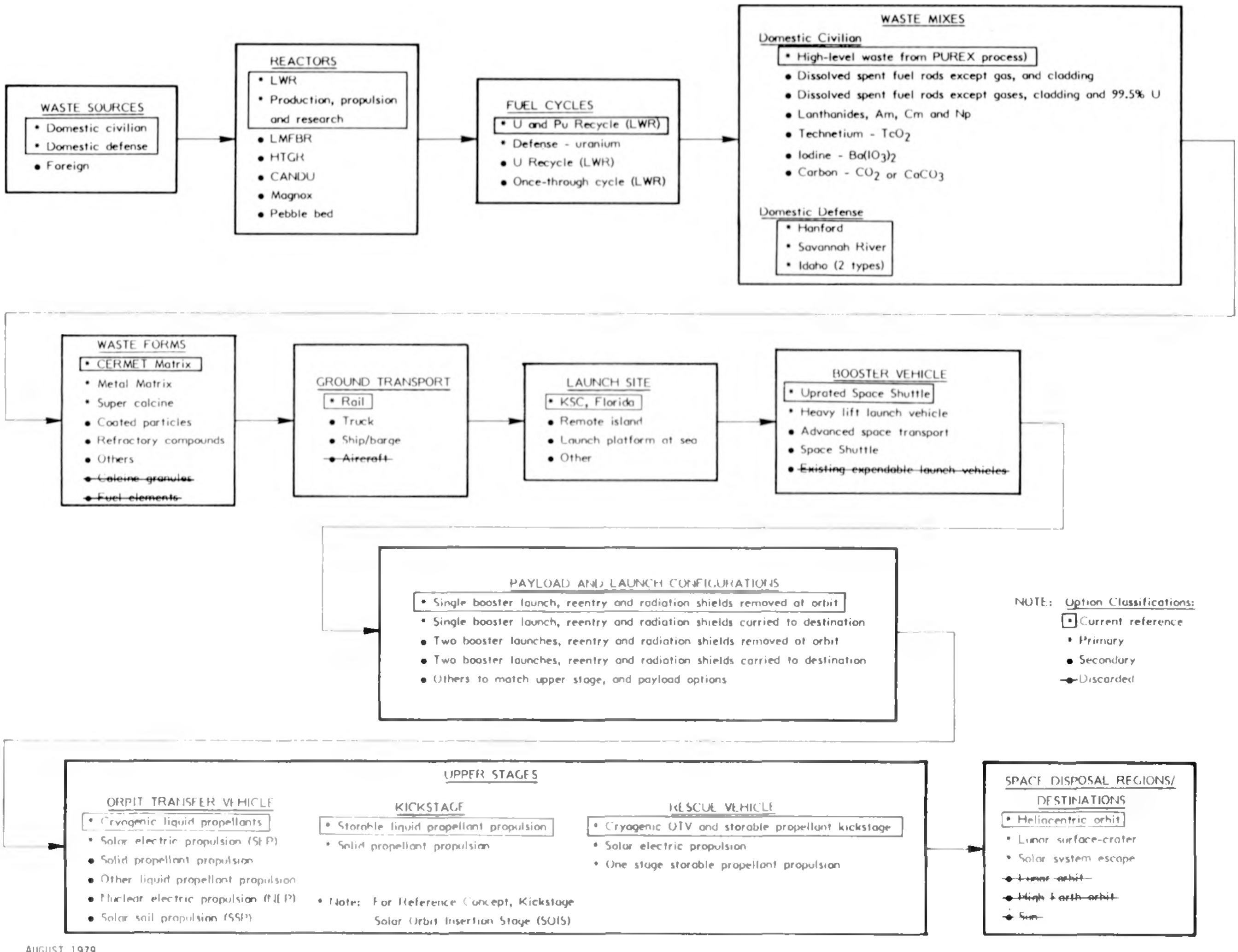

FIGURE 3. MAJOR OPTIONS FOR SPACE DISPOSAL OF NUCLEAR WASTE

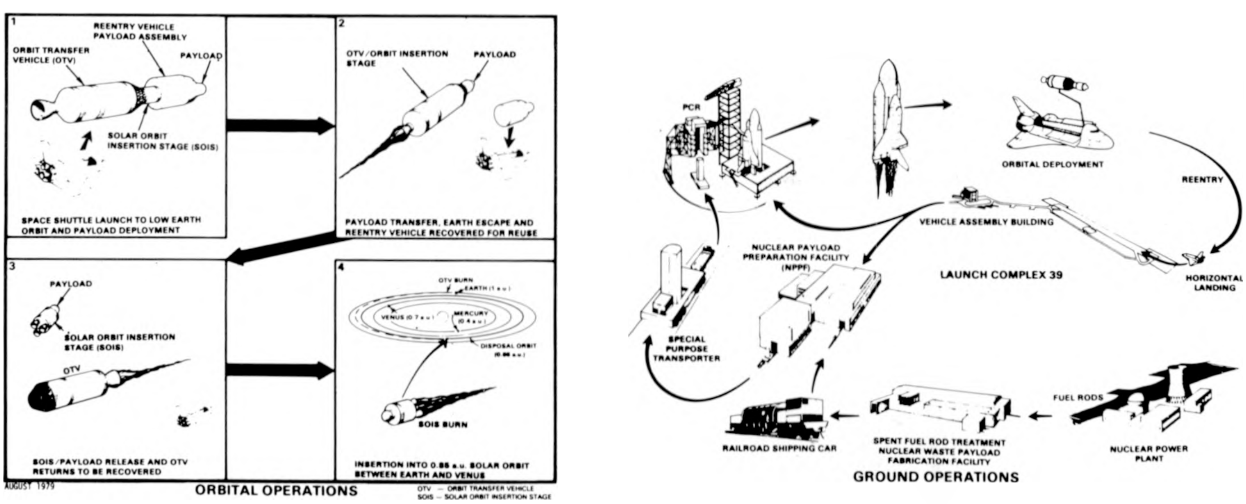

FIGURE 4. GROUND AND SPACE OPERATIONS FOR REFERENCE SPACE DISPOSAL CONCEPT

COMPARING SOCIAL AND INSTITUTIONAL ASPECTS OF ALTERNATIVE WASTE-DISPOSAL TECHNOLOGIES

A. H. Schilling
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The Battelle Human Affairs Research Centers (HARC) are examining social and institutional aspects of alternative waste-disposal technologies. This work will lead to comparison of various alternative technologies on social and institutional grounds. More importantly, the data on these nontechnical aspects can be integrated with the technical information to allow a fuller assessment of the alternatives and thus better decisions about them. This work is intended to provide a social science complement to an assessment being undertaken for NWTS/ONWI by Bechtel National, Inc.

The technologies considered are: deep hole, rock melt, space, seabed, and transmutation. These five will, in turn, be compared with the base case, mined geologic repositories.

Conceptually, the HARC work fills in the cells of a matrix whose rows are the six alternatives and whose columns relate to various social and institutional aspects. These include: socioeconomic impacts (as generally understood in preparing EIS's), legal barriers, organizational capabilities and interests, and political support. This matrix can then, at least conceptually, be matched with a companion matrix having columns which relate to cost, feasibility, environmental impacts, and other technical aspects of the alternatives. This will permit a complete evaluation of the relative merits of the alternatives. A starting point for this matrix is a similar matrix presented in Section 4 of the "Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste."

There is another perspective in the analysis. Ordinarily, the emphasis on policy analysis, including technology assessment, is on the outcomes—in this case, which technologies to study further, which to implement, and which to reject. Much less attention goes to procedures for implementing the policy. The kinds of social and institutional aspects examined in the HARC work are prominent in implementation.

As a first step at implementation analysis, the social and institutional matrix might be expanded to include cost and feasibility information, borrowing from previous studies and from the technology assessment. Alternatives could, most simply, be compared by looking at the expanded matrix. One might move into a more quantitative mode by scoring each alternative (row) on each attribute (column), although reliable quantitative scores for legal, organizational, and political aspects are difficult to define. If one attempts such a scoring, it then becomes necessary to think about the comparative importance of the attributes, i.e., to weight them. Since weighting (e.g., deciding a legal problem is twice as important as a cost problem) is essentially in exercise in value judgment, it then becomes necessary to take the analysis beyond the technical community to either policymakers, special interest groups, or the public at large.

The matrix comparison discussed so far lends itself to preliminary judgments among alternatives. It serves to force consideration of the important aspects of a decision. While it is sometimes possible to introduce quantitative rigor into the analysis, care needs to be taken to balance the costs and benefits of doing so.

A further step in implementation analysis looks at institutional constraints, prior to implementing alternatives. A useful tool in doing this is the mapping of the implementation process. The mapping exercise is conceptually similar to a dynamic programming technique known as the "transportation problem". It is also not too far removed in general outlook from PERT charting. The idea is to work backward from where you want to be to where you are, taking note of the important points along the way. Here, one would map with legal, organizational, and political factors primarily in mind.

There are a number of analytical complications. First, the six alternatives are not independent. Space disposal, for example, is not feasible for disposal of all commercial nuclear waste. Indeed, it would be an expensive way to dispose of spent fuel. There will need to be other waste-disposal systems to complement the space alternative. The same is likely to be true for transmutation. If the complement for these alternatives is mined geologic repositories, the social and institutional comparisons become complex. It is difficult to develop sufficiently reliable quantitative scores for comparative assessment in these areas. Hence, tradeoffs between the implementation of repositories only ("geological disposal") and the implementation of somewhat different (e.g., size, number, long-term risk, etc.) repositories combined with space shots ("space disposal") are difficult to assess.

Another problem is uncertainty. For example, socioeconomic impacts are generally quite site-specific. Space disposal utilizing shuttle launches from the Kennedy Space Center would likely involve fewers such impacts because the facilities and site and community are already in place. Sending the waste off from a new launch site developed for the space-disposal mission could, on the other hand, entail major impacts. One cannot specify impacts well without choosing a site. In addition, there are socioeconomic-impact uncertainties about the technologies themselves, about their attributes, and about matches among the two. Such uncertainties must be kept in mind in making assessments. Indeed the uncertainties of the assessments should be carefully quantified, because decisions about which uncertainties to reduce may be initially more important than which technologies to choose.

AN ASSESSMENT OF ALTERNATIVE WASTE-DISPOSAL CONCEPTS

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Bechtel National, Inc.

Background

For many years, research and development have been conducted by government and industry throughout the world to demonstrate acceptable methods for the management and permanent disposal of radioactive wastes produced by commercial nuclear power plants. In the United States, the Department of Energy (DOE), its predecessor organizations, and various contractors have studied and developed many disposal concepts in order to ensure that safe isolation can be provided at acceptable costs to society. Disposal concepts which have previously been assessed and compared include: mined geologic, seabed, subseabed, space, partitioning and transmutation, chemical resynthesis, very deep hole, rock melting, island, ice sheet, and reverse well.[1-4] Among the alternative concepts identified for further consideration by the IRG[5] were mined geologic, subseabed, very deep hole, and space. No assessments were made. The Alternative Waste Disposal Concepts Study (AWDCS), being conducted by Bechtel for NWTS/ONWI, represents a continuing effort in definition and assessment of these concepts and two others: rock melting and partitioning and transmutation. The study began in April, 1979, and the final report is scheduled to be completed at the end of November, 1979.

Study Cases

The AWDCS program is evaluating a total of nine processing and disposal concepts. These are combined, as appropriate, into selected study cases. The three processing concepts are:

- Encapsulation in steel canisters of intact spent fuel assemblies to prepare them for final disposal, as a part of a once-through uranium oxide fuel cycle
- Reprocessing of spent mixed-oxide fuel (with accompanying recycle of uranium and plutonium) in preparation for the final disposal of encapsulated high-level waste (HLW) and other reprocessing waste
- Reprocessing of spent mixed-oxide fuel as above, with the added step of partitioning specific long-lived radioactive elements for separate disposal.

The six disposal concepts are:

- Deep-mined geologic repository in salt (reference concept)
- Very deep hole (about 10,000 meters)
- Subseabed
- Solar orbit

- Transmutation by radioactive bombardment to less hazardous nuclides
- Placement of high-heat-producing waste in a deep rock cavity and allowing the heat to melt the rock, forming a dispersion-resistant waste/rock matrix when the resultant waste and rock solidify.

There are 18 possible study cases obtainable by simply combining the three processing concepts with the six disposal concepts. Indeed, a preliminary review indicated that even more than 18 cases are possible if the study were to consider variations identified within some of the concepts. Because of time and budget constraints, however, it was necessary to limit the number of cases to those considered most promising at this time.

The partitioning and transmutation concepts were reviewed early in the study. Both were set aside from further consideration in this study because of fundamental uncertainties regarding their feasibility and/or the immaturity of their technology. They are both judged to be impractical within the time frame in which disposal options are expected to be developed.

Space disposal of simply encapsulated spent fuel elements was also eliminated from further consideration because of the extremely large number of shuttle flights that would be required (>1,000 flights per year for a 5000-MTHM/year fuel-cycle system). Space disposal of HLW was included in the study.

Spent-fuel emplacement in a rock melting cavity was eliminated because of the need for a liquid high-level waste stream in the implementation of this concept.

Eight processing/disposal concept combinations (cases) are thus defined and assessed in this study, as indicated in Table 1.

Each study case includes processing, storage, transportation, and disposal of all waste streams generated in the back end of the fuel cycle. The cases also include the transportation to recycle facilities for those cases involving the recovery of uranium and plutonium. The scope of the spent-fuel disposal cases in this study is shown in Figures 1 through 8. The facilities and transportation steps within the dotted lines are those included in the study.

Assessment Approach

Following a preliminary definition of the study cases and concepts identified above, the cases are analyzed and compared in terms of the following assessment criteria:

- Radiological impact
- Nonradiological environmental impact
- Degree of development
- Resource consumption
- Safeguards
- Economics.

A systems study approach similar to that used in the Spent-Fuel Disposal Study[6], reported on earlier, is used in making the evaluations and comparisons of the study cases. For each assessment criterion, the eight study cases are evaluated on a qualitative or quantitative basis as described below, taking into account the relative uncertainties of the disposal concepts defined. No attempt is made in this study to develop weighting or preference factors to enable the assessments for the six criteria to be combined.

The **radiological impact** assessment develops, for each case, approximate dose values for the short- and long-term effects for occupational and nonoccupational segments of the population. The dose values are estimated on the bases of established data for similar facilities and operations.

The **nonradiological environmental impact** assessment examines the physical, biological, and social effects of constructing, operating, and decommissioning the facilities for each case. The assessment compares cases on the basis of the absolute magnitude of project impacts; site-specific effects are not considered.

The **degree of development** assessment analyzes the status of the technology for each case in terms of the following attributes or considerations: emplacement methods, emplacement medium, waste form, waste containment, and facilities.

The **resource consumption** assessment evaluates the materials, labor, land, and capital required for case implementation. These data are developed from published reports as well as from Bechtel in-house sources.

The ability of a **safeguards** system to provide and maintain a secure environment for the nuclear materials for each of the eight study cases is investigated. The potential risk to the public from sabotage and theft of nuclear materials is assessed.

The **economic** assessment produces a levelized average cost in dollars per metric ton of heavy metal processed for each case. The values of recovered uranium and plutonium processed are applied in those cases where applicable.

Status

At the time this summary was prepared, the concept and case definitions were nearly complete for the AWDCS, and the assessments were under way. A draft report of the study was scheduled to be submitted to ONWI by September 30, 1979, and the final report by November 30, 1979. No further work for FY 1980 under this subcontract has been identified to date.

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TABLE 1. CASE IDENTIFICATION IN ALTERNATIVE WASTE-DISPOSAL CONCEPTS STUDY

	Processing Concepts		
Disposal Concepts	Spent-fuel Encapsulation	Chemical Processing and Recycle	Chemical Processing Partitioning And Recycle
Mined geologic repository	A	D	Eliminated
Very deep hole	B	E	Eliminated
Rock melting	Eliminated	F	Eliminated
Subseabed	C	G	Eliminated
Space	Eliminated	H	Eliminated
Transmutation	Eliminated	Eliminated	Eliminated

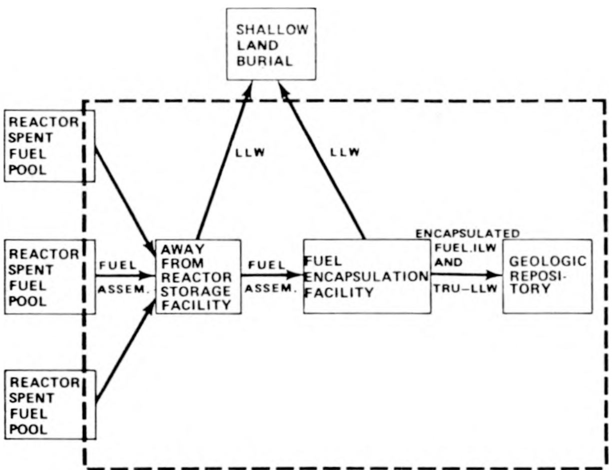

FIGURE 1. FUEL-CYCLE DIAGRAM – CASE A: SPENT FUEL IN A MINED GEOLOGIC REPOSITORY

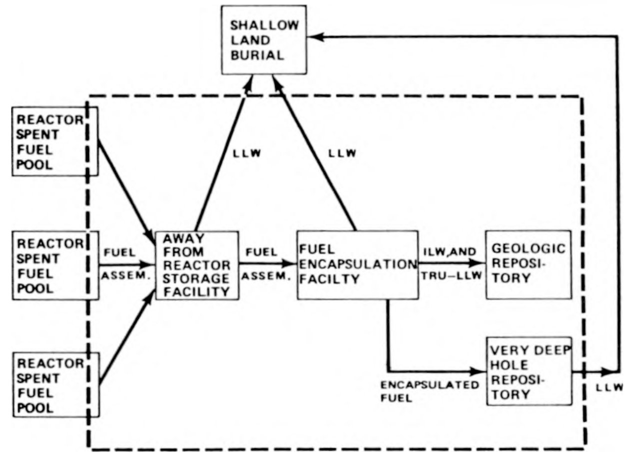

FIGURE 2. FUEL-CYCLE DIAGRAM – CASE B: SPENT FUEL IN A VERY DEEP HOLE

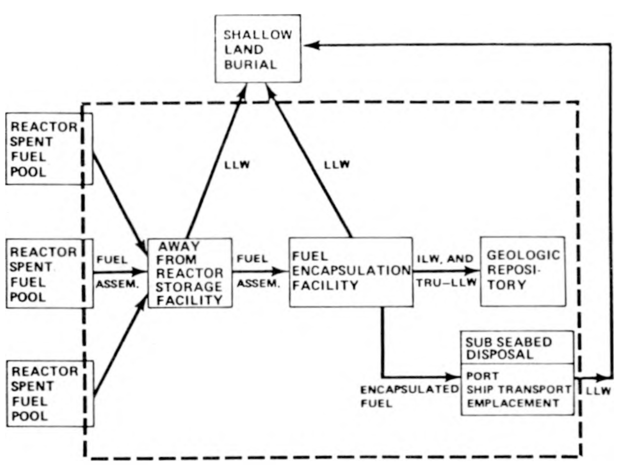

FIGURE 3. FUEL-CYCLE DIAGRAM – CASE C: SPENT FUEL IN THE SUBSEABED

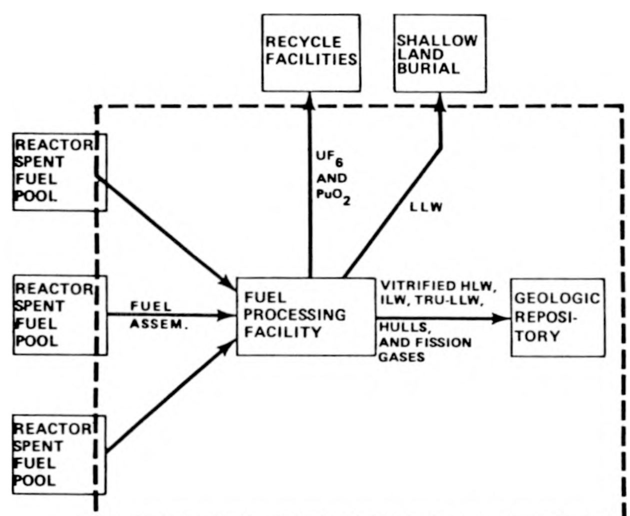

FIGURE 4. FUEL-CYCLE DIAGRAM – CASE D: VITRIFIED HLW IN A MINED GEOLOGIC REPOSITORY

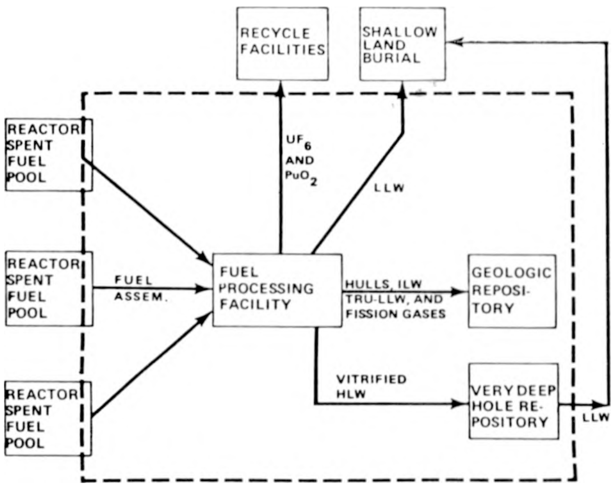

FIGURE 5. FUEL-CYCLE DIAGRAM – CASE E: VITRIFIED HLW IN A VERY DEEP HOLE

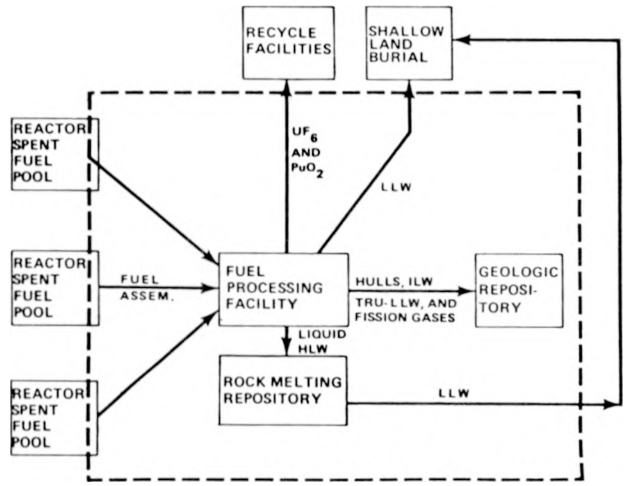

FIGURE 6. FUEL-CYCLE DIAGRAM – CASE F: LIQUID HLW IN A ROCK MELTING REPOSITORY

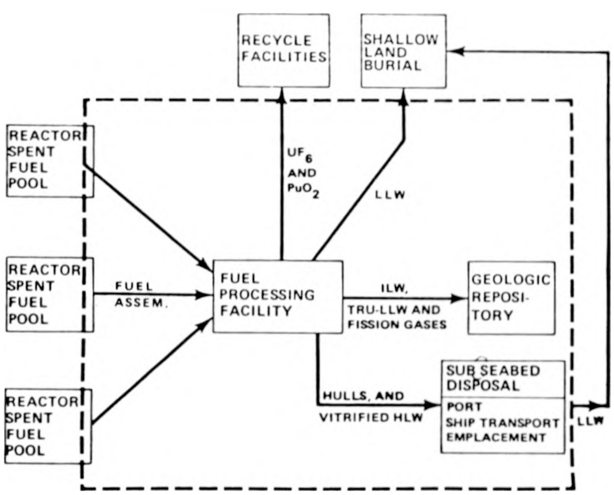

FIGURE 7. FUEL-CYCLE DIAGRAM – CASE G: HULLS AND VITRIFIED HLW IN THE SUBSEABED

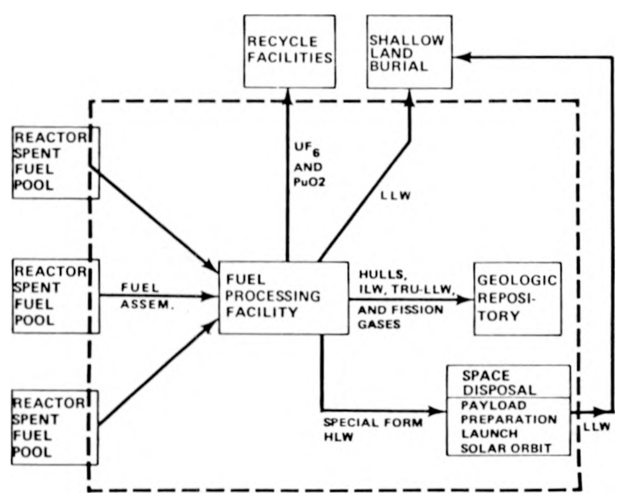

FIGURE 8. FUEL-CYCLE DIAGRAM – CASE H: SPECIAL FORM HLW IN SPACE

CRITERIA COORDINATION

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Office of Nuclear Waste Isolation

The objective of criteria coordination is to assure consistency throughout the NWTS Program in the standards of measurement applied in defining, developing, and evaluating a repository system for the disposal of radioactive wastes. A supplementary goal is to minimize overlap and duplication in criteria development efforts. For this purpose ONWI is developing, and attempting to achieve consensus on, criteria that can be applied throughout the NWTS program. The latter is being accomplished by means of meetings of NWTS Program participants that provide for the review and discussion of criteria development efforts and by the formal review and critique of draft criteria documents.

ONWI criteria development efforts fall in three areas—the waste package, the site, and the repository. Major efforts have been underway during the past year to produce criteria documents in two of the areas—waste-package, criteria and site-qualification criteria. These are general criteria in that they are not related to a specific geologic medium, site, or project. Criteria specific to and responsive to the technical and socioeconomic characteristics of a specific site or project will be developed by each NWTS repository projects must be responsive to these criteria, these projects have been factored into the criteria review, comment, and approval chain. In the case of waste-package criteria—because of their broader impact across the waste-management system—the waste processing, storage, and transportation elements of the national waste management effort have been included in this chain.

The early waste-package criteria coordination effort of ONWI led to a significant accomplishment. This effort was initiated to assure that transuranic waste-package acceptance criteria being developed for the NWTS Program were compatible with those being developed for WIPP. The Waste-Acceptance Criteria (WAC) Developers Group was formed for this purpose and provided a forum for waste management system-wide input to and discussion of criteria and their bases. One of the early conclusions arrived at by this group was that criteria defining general system and repository performance requirements were needed that would be applicable to all repository systems and all waste types. A working group was formed to develop proposed criteria and involved representatives from ONWI contractors working on criteria (BCL, RHO, and PNL), as well as representatives from the WIPP Project (*W* and SL) and from SRL. The output was a major source for the document, "NWTS Program Criteria for the Geologic Disposal of Nuclear Wastes—General Program Policies and Criteria". It includes top-level criteria and direction for NWTS-wide application in such areas as sources of waste for disposal, the level of safety that must be achieved in the isolation of nuclear waste, licensing requirements, the application of multiple barriers to waste isolation, retrievability, and operaional safety and economics.

Waste-package efforts in FY 1980 will emphasize the development of specifications that respond to the criteria. The major criteria effort will be directed toward definition of criteria for repository design and operation. In addition, a criteria management system will be established to control, approve, and record all criteria as they are developed and as changes may become necessary.

DEVELOPMENT OF WASTE-ACCEPTANCE CRITERIA

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R. E. Smith, M. S. Bensky, and E. L. Moore
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Objective

The objective of this study is to establish proposed criteria by which waste packages, including the contained waste form, can be determined to be acceptable for handling and disposal at a deep-mined geologic repository.

Scope

The wastes for which acceptance criteria are required are those from the LWR fuel cycle and include: high-level waste from reprocessing operations; spent fuel from the once-through fuel cycle; cladding hulls; remote-handled transuranic waste; contact-handled transuranic waste, including waste from decontamination and decommissioning operations; and fission-product-gas wastes. The intent is to establish acceptance criteria which will be compatible with both commercial and defense wastes, and also with criteria that may be appled to LMFBR and HTGR wastes. The criteria are general, since they are not restricted to a specific repository or geologic medium. Individual repository projects will develop criteria and specifications specific to the technical requirements of a given site.

Role and Support of NWTS/ONWI

Acceptance criteria represent a fundamental starting point in the development of any repository or waste-disposal system. In establishing these criteria, it is necessary to consider all elements of the disposal system (see Figure 1) in order to identify constraints which must be imposed on the waste for acceptability in the system. These constraints are defined through analysis of the waste emplacement and isolation requirements. The analysis results in the description of the waste-package characteristics and properties which must be provided by appropriate control of treatment and packaging operations to make it acceptable for disposal.

Background

This effort has grown out of two studies that were initiated by the Office of Waste Isolation: a container-criteria effort and spent-fuel packaging study by Rockwell Hanford Operations (RHO), begun in late FY 1976, which resulted in the preparation of preliminary packaging criteria[1]; and a study which was contracted to Battelle's Columbus Laboratories (BCL) in FY 1978 to establish waste (form) acceptance criteria. These programs became the responsibility of ONWI in July, 1978.

Shortly after ONWI assumed responsibility for these programs, it was recognized that closer coordination was required to integrate these and other activities related to waste-management system development. The responsibility of the two contractors was more clearly delineated, and the need for coordinating activities with Waste Isolation Pilot Plant (WIPP) efforts in this area were initiated to ensure compatibility between the two sets of acceptance criteria was recognized. A Criteria Developers Group was formed to provide a mechanism for assembling representatives from various elements of the NWTS Program to discuss criteria development, the impact of various elements of the waste-management system on criteria, and, specifically, to promote compatibility between criteria for commercial waste and for defense waste. During the early stages of this task, the Criteria Developers Group realized that, while acceptance criteria for specific waste types might differ, higher level criteria appropriate to all waste sources and types could be formulated. It was also recognized that the waste disposal objective could be achieved by different means. This group identified the need for general performance criteria which would provide the context, direction, and interfaces for the development of more specific waste-package performance, acceptance, and design criteria.

The efforts of the working group resulted in the preparation of a draft document, "Criteria for

Geologic Disposal of Nuclear Wastes, General Program Policies and Criteria",[2] that describes the general performance criteria for the waste disposal system as they relate to geologic disposal. A logic framework for criteria development, referred to as a criteria hierarchy, also was presented. This draft document served to provide guidance for development of the proposed waste-package criteria.

Waste-Package Performance and Acceptance Criteria

Concurrent with the above criteria activity, the development of waste-package performance and acceptance criteria was pursued by RHO and BCL, with the responsibilities indicated previously. These contractors have prepared a draft document, "Criteria for Geologic Disposal of Nuclear Wastes, Interim Waste Package Performance and Acceptance Criteria"[3]. The package performance criteria document provides a discussion of each criterion, including the present quantification of the criteria acceptance criteria identify and, as data are available, recommend limits or means of identifying limits for characteristics of the waste package (see Table 1) which must be controlled to achieve the required performance. Also included in the interim criteria report is a series of decision logic diagrams which describe the questions that must be answered and identify the research efforts required to quantify the final criteria and to set specifications.

Plans for FY 1980

The emphasis in FY 1980 will be on studies to develop and document bases for waste-package specifications on the basis of the interim waste-package criteria developed in this effort.

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(2) Draft in review.

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TABLE 1. FUNDAMENTAL AND DERIVED CHARACTERISTICS WHICH MAY REQUIRE CRITERIA

- Nuclear Characteristics
  - Fundamental Characteristics
    - Nuclide Concentration
    - Fissile Concentration
    - Lattice Geometry
  - Derived Characteristics
    - Fissile Load
    - Surface Contamination
    - Surface Dose Rate
    - Shielding
    - Radiation Damage (Stored Energy)
    - Specific Activity
- Chemical Characteristics
  - Fundamental Characteristics
    - Chemical Composition and Form
    - Atmospheric Composition
  - Derived Characteristics
    - Gas Generation Rate
    - Flash Point
    - Ignition Temperature
    - Corrosion Rate
    - Phase Change
    - Leach Rate
- Physical/Mechanical Characteristics
  - Fundamental Characteristics
    - Canister Material Properties
    - Package Geometry
    - Void Volume
    - Emplacement Configuration
  - Derived Characteristics
    - Structural Integrity
    - Weight
    - Surface Area
    - Internal Pressure and Overpressure
    - Leak Rate
    - Stress Cracking
    - Particle Size
    - Dispersibility
- Thermal Characteristics
  - Fundamental Characteristics
    - Thermal Conductance
  - Derived Characteristics
    - Surface Temperature
    - Bulk Temperature
    - Thermal Load

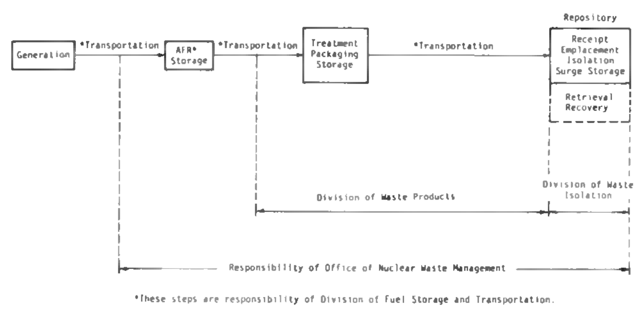

FIGURE 1. WASTE MANAGEMENT SYSTEM

EVALUATION OF THE REGIONAL REPOSITORY CONCEPT FOR NUCLEAR WASTE DISPOSAL

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Previous planning associated with the NWTS Program assumed the selection of a few large national repositories for commercially generated nuclear waste. The Interagency Review Group[1] recommended an alternative approach, which proposes the construction of multiple, smaller repositories sited on a regional basis. Study has begun within the NWTS/ONWI Systems Analysis Program to evaluate the concept of regional repositories for nuclear waste disposal and to assess the potential impact on the NWTS Program. Elements of the study have included the definition of possible regions, predictions of the total and annual inventories associated with each region for different nuclear capacity scenarios, determination of the accompanying transportation costs and population exposures, and relative assessments of the effects of this concept on repository facility and operating costs.

Various groupings of states and utilities have been investigated as possible candidate regions. These candidates encompass existing governmental regions, power pools, conventional geographic regions, and combinations yielding relatively equal nuclear electrical generation capacity. The particular candidates or groupings that have been examined are given below; the number of regions associated with each is given in parentheses.

- Single U.S. (1)
- Nuclear Regulatory Commission (5)
- Federal Energy Regulatory Commission (5)
- National Electric Reliability Councils (9)
- Equal nuclear generation capacity (3,4,5,7,9)
- Geographic (5,9,12)

The focus of the study has been the evaluation of the regional repository option as a concept, and little effort has been devoted to the determination of specific repository locations within the various regions. For evaluation purposes, a repository has been placed within each region at a reasonably centralized location that contains one of the candidate geologies presently identified in the draft GEIS.[2]

Investigations of the regional concept have been performed on the basis of spent fuel from U.S. commercial generating plants only, for four nuclear-capacity scenarios. The first scenario yields 186 GW(e) of installed capacity in the U.S. by the year 2000 and corresponds to those plants currently operating, under construction, or announced.[3] Scenarios yielding 325 GW(e) and 400 GW(e) in the year 2000 have also been investigated; the latter corresponds to the reference GEIS scenario. These three scenarios consider no additional plants beyond the year 2000, but the total spent-fuel discharges throughout the plant lives are considered. With a 40-year plant life and a minimum 7-year cooling prior to disposal, the analysis period extends to the year 2047. Detailed transportation characteristics up to the year 2010 have been evaluated separately by Oak Ridge National Laboratory[4], using a scenario reaching 325 GW(e) in 2000 and increasing to 419 GW(e) by 2008.

The use of multiple regions, each with its own repository, obviously reduces the inventories and receiving rates of the individual repositories as compared to a single large repository. Table 1 shows the relative inventories associated with each region for several regional examples, based on the 400-GW(e) GEIS scenario. Maximum receiving rates are similarly reduced.

The regional-repository concept would be expected to exhibit distinct advantages over the single-repository approach in the area of transportation requirements and associated population risks. Regional transportation characteristics have been investigated, and the expected advantages have been confirmed. Figure 1 illustrates the reduction in unit transportation costs as the number of regions increases. Total transportation costs for the selected examples and the four capacity scenarios are given in Table 2. Table 3 shows the sizes of the cask fleets required to support selected regional examples. These results indicate a significant reduction in transportation costs and cask-fleet requirements for as few as three regions, and they show that, even for different examples of five regions, reductions by about a factor of two can occur compared with a single large repository. Population radiation exposure shows a similar reduction, as illustrated by Figure 2.

Assessment of the influence of the regional concept on facility and operating costs is difficult at present. Available data do not allow reasonable scaling of these costs as a function of total and annual inventories, geology, timing, and other economic ground rules. Initial evaluations that may possibly bracket the relative influence of the regional concept on facility costs have been performed using generally accepted scaling conventions and available data. Facility costs for a reference repository in bedded salt with a capacity of approximately 103,000 mtU of spent fuel are estimated to total about $607 million. Figure 3 illustrates the potential effect of the regional concept on facility costs, based on scaling by the ratio of individual repository capacity to the 0.5 and 0.7 power. Lifetime operating costs associated with the reference repository have been estimated to be $738 million. Investigations to assess the effect of the regional concept on operating costs are currently in progress.

The present study includes continued investigation of various technical and economic aspects of the regional concept. Factors such as licensing and environmental impact will be examined, and the study will also assess more fully the potential impact of the regional concept on the current NWTS Program Plan. The regional concept would require significant, parallel efforts in exploration and development of alternate geologies and may require considerable standardization of repository design and licensing. Coordination of efforts with regulatory agencies and state/local planning groups will also become more important for successful implementation of the regional concept. General plans for FY 1980 include more-detailed economic evaluation, investigation of the effects of other interim storage philosophies, including away-from-reactor storage, inclusion of other waste types and sources, consideration of occupational exposure and ALARA effects, study of safeguards and security, and the incorporation of social, political, and regulatory considerations.

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TABLE 3. MAXIMUM SHIPPING-CASK FLEET REQUIREMENTS FOR REGIONAL REPOSITORIES (ORNL CAPACITY SCENARIO)[a]

Regional Example	Rail Casks	Truck Casks
U. S. Single		
Gulf Interior	134	75
Central IL	113	63
Pacific NW	227	118
Three, equal capacity	95	61
Five, NRC	69	51
Five, equal capacity	76	53
Nine, NERC	66	59

[a] Capacity scenario yields 325 GW(e) in the year 2000 increasing to 419 GW(e) by 2008. End of the analysis period is 2010; total shipment inventory is 145,600 mtU. Shipping casks are used within the individual regions only; no interrregional shipments. Rail casks (NLI 10/24) are used for plants with rail access; truck casks (NLI 1/2) are used otherwise. An average rail/truck split of about 80%/20% results.

TABLE 1. REGIONAL DISTRIBUTION OF POTENTIAL CUMULATIVE SPENT-FUEL INVENTORY FOR GEIS 400-GW(e) SCENARIO[a]

Regional Example	Percent of Total Inventory	Regional Example	Percent of Total Inventory
U. S. Single	100	National Electric Reliability Council Regions	
Nuclear Regulatory Commission Regions		1. ECAR	13.8
		2. ERCOT	5.7
1. Region I	20.9	3. MAAC	8.8
2. Region II	31.5	4. NPCC	11.0
3. Region III	23.7	5. MARCA	3.1
4. Region IV	12.0	6. MAIN	9.7
5. Region V	11.9	7. SERC	26.6
		8. SPP	7.5
Federal Energy Regulatory Commission Regions		9. WSCC	13.7
		Nine Geographic Regions	
1. Northeast	25.5	1. New England	6.1
2. Southeast	27.2	2. Middle Atlantic	12.7
3. Midwest	20.9	3. South Atlantic	18.6
4. South Central	13.8	4. East North Central	21.1
5. West	12.5	5. West North Central	4.4
Five Geographic Regions		6. East South Central	12.3
1. Northeast	20.9	7. West South Central	11.1
2. South	31.5	8. Mountain	4.7
3. Midwest	25.5	9. Pacific	9.0
4. Southwest	11.2		
5. West	10.8		

[a] Total cumulative spent-fuel inventory is 398,500 mtU. Total capacity scenario yields 400 GW(e) in the year 2000 (366 plants). Expansion plants located at existing sites and arbitrary new sites with at least one plant per state. No new plants after the year 2000. Inventory includes all spent fuel discharged during the 40-year plant life.

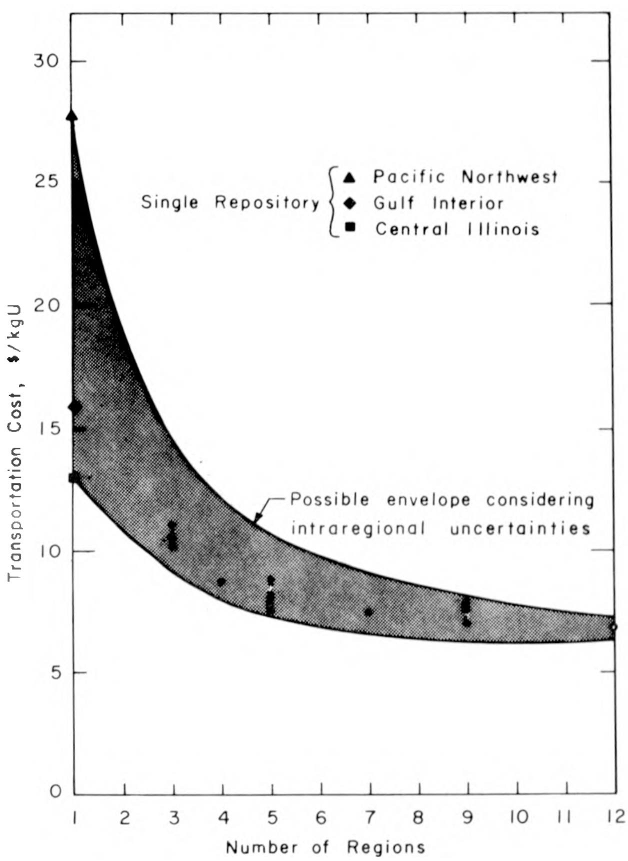

FIGURE 1. UNIT TRANSPORTATION COSTS FOR REGIONAL REPOSITORIES

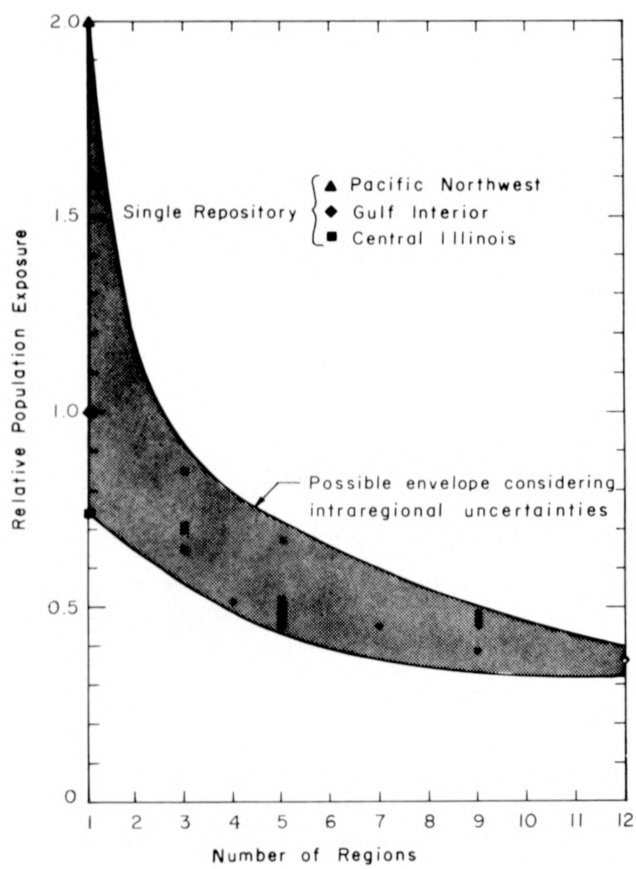

FIGURE 2. RELATIVE POPULATION RADIATION EXPOSURE FOR REGIONAL REPOSITORIES

TABLE 2. TOTAL TRANSPORTATION COSTS FOR SHIPMENT OF SPENT FUEL TO REGIONAL REPOSITORIES

Regional Example	Costs (10^6\$) for Capacity Scenario:[a] 186 GW(e)	325 GW(e)	400 GW(e)	ORNL
1. Single U.S.	2950	4790	5800	—
Gulf Interior	—	—	—	2305
Central IL[b]	—	—	—	1889
Pacific NW	—	—	—	3994
2. Three, equal capacity	—	—	4054	1505
3. Four, equal capacity	—	—	3464	—
4. Five, NRC	1484	2523	3123	1109
5. Five, FERC	1476	—	3060	—
6. Five, geographic	1518	—	3184	—
7. Five, equal capacity	—	—	3217	1197
8. Five, equal capacity[c]	—	—	3509	—
9. Seven, equal capacity	—	—	2978	—
10. Nine, NERC	1447	2414	2961	1056
11. Nine, geographic	—	—	2831	—
12. Nine, equal capacity	—	—	2996	—
13. Twelve, geographic	—	—	2674	—
Total Inventory (10^3mtU)	191.2	326.5	398.5	145.6

(a) Shipments are within individual regions only, directly from reactor to repository. Spent fuel is stored at reactor site until regional repository opens and shipped following a minimum 7-year cooling. Backlog of older fuel is shipped uniformly between the years 2000 and 2010. ORNL data are from Reference (4).
(b) Hypothetical location with minimum mtU-miles for total transport.
(c) Same regions as Item 7 with different repository locations.

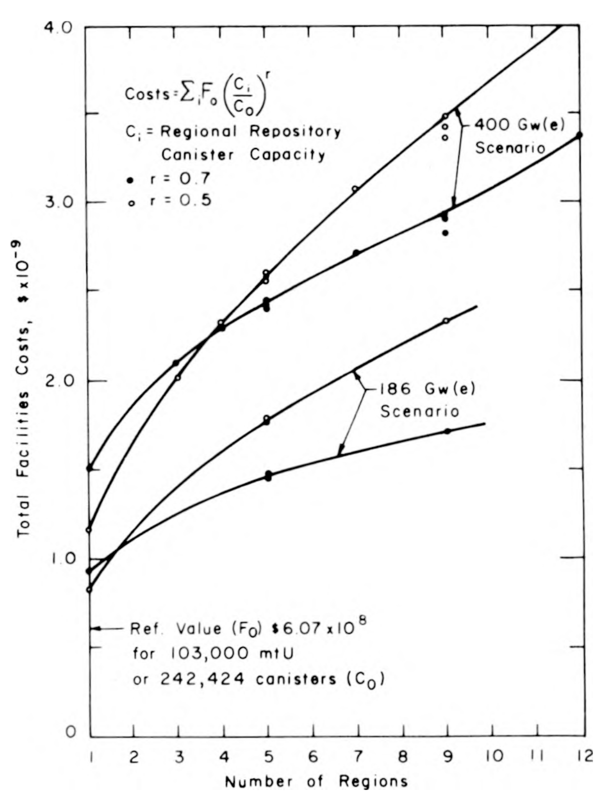

FIGURE 3. POTENTIAL TOTAL FACILITY COSTS FOR REGIONAL REPOSITORIES

OPTIONS FOR RETRIEVAL AND RECOVERY FROM A REPOSITORY

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The Retrieval Options Study, directed by NWTS/ONWI, had as its objective the development of a basis for the design of the retrieval process and equipment used in a mined-repository waste-isolation system. The study began in March, 1979, and was scheduled for completion by the end of September, 1979. The scope of work required by Kaiser Engineers, Inc., is to systematically investigate and evaluate several retrieval options using measures of cost, safety, technical feasibility, difficulty of retrieval, probability of intact retrieval, safeguards, monitoring, criticality, and licensability. Systems analysis was required to provide an overview of interdisciplinary analyses performed in broad problem areas relating to waste retrieval, and to identify gaps in knowledge or technology requiring further engineering evaluation to establish an acceptable projection of repository performance.

The study was divided into 3 separate tasks: (1) to define retrievability from open storage rooms and recoverability from backfilled storage rooms, (2) to synthesize and analyze emplacement/retrieval options, and (3) to prepare and present information in a form allowing the evaluation of a specific retrieval option.

The definitions of retrievability and recoverability that were developed are based on the operational effort required in the mine. Class designations were also made to define the state of the mine and waste package at the time of retrieval/recovery. A broad range of conditions was considered. In the least difficult operation, the storage room is open and the waste package is free in the deposit hole; however, operational requirements include a ventilation period to lower mine temperatures, so as to permit the entry of personnel, cleanup of passages and rooms, and the subsequent retrieval of the waste. In the most difficult operation, the mine is completely backfilled and decommissioned and waste packages may not be structurally sound. In all cases, retrieval is influenced by the type of rock, condition of the mine, time period of waste storage, mine cooling characteristics, the condition of the waste package, the requirements for specially designed recovery equipment, and the rate of recovery.

Table 1 summarizes 32 base-case storage arrangements that were established in order to synthesize available retrieval options. These arrangements represent four rock types (salt, basalt, granite, and shale); two areal thermal loadings (kw/acre); two waste package forms (high-level waste canister and spent-fuel canister); and two canister orientations (vertical and horizontal). Figure 1 presents the master mine plan adopted for all base cases; this plan was originally developed in Kaiser Engineers' conceptual design of the NWTS repository in bedded salt.

Thermal analyses were performed to develop technically feasible storage arrangements in each case and to establish the associated thermal conditions during representative retrieval and recovery operations. Design criteria that had been developed for each rock type in the overall NWTS/ONWI Program were used for guidance; the criteria included: design areal thermal loadings, maximum allowable temperatures for the waste forms, and limitation on the percentage of repository area mined. A storage arrangement was selected on the basis of rock temperatures calculated for candidate vertical storage arrangements at the design areal thermal loading. Rock temperatures were also calculated for arrangements representing two extremes of halving the design areal thermal loading in the selected arrangement: removing half of the canisters in each storage room, and, alternatively, deleting every other storage room. However, the analysis of retrieval options was considered only for the selected storage arrangement which meets the design areal thermal loading and for the most favorable of the arrangements established for the reduced areal thermal loading.

Each of the defined retrieval and recovery conditions was analyzed for each base case. Major analyses were performed in the technical areas of mine engineering and waste handling. Analyses were also made in the areas of safety and licensing, nuclear engineering, instrumentation and controls, and metallurgy.

Mining procedures were established for each option that will allow the entry of personnel and equipment into the storage room to effect retrieval. This included the specification of mine equipment operations, personnel requirements, ventilation requirements, and additional capital cost items needed for retrieval.

Waste-handling procedures were established to suit the storage condition at the time of retrieval. This included the specification of repository facilities and equipment required to achieve an acceptable recovery rate. In some extreme cases, highly specialized recovery techniques implemented with specially designed machinery were found mandatory. Moreover, extensive modifications to waste-handling areas, both in the mine and in the surface facility, were found necessary.

A systems analysis methodology was developed for the preparation and presentation of the technical information developed for each retrieval option. Each option was defined by a set of parameters which describe the associated base case, the condition of the facility at the time of the retrieval, and the details of the waste-package emplacement in the storage hole. The influence of each parameter on retrieval was then evaluated using a rating system to characterize an option in each measurement area; i.e., cost, safety, technical feasibility, difficulty of retrieval, etc. The results obtained provide a data base from which the overall worth of a retrieval option can be assessed.

The task of nuclear waste retrieval from a mined repository was found to be a unique engineering challenge. Moreover, the magnitude of the task is such that all available options must be carefully studied so as to provide the safest and most practical repository operation.

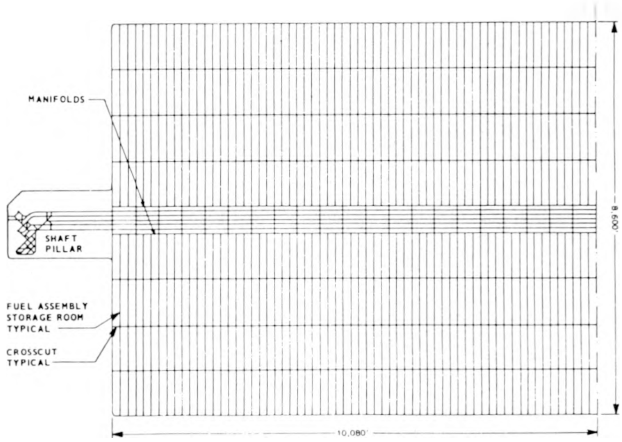

FIGURE 1. MASTER MINE PLAN

TABLE 1. BASE-CASE STORAGE ARRANGEMENTS

Storage Medium	Thermal Loading, kw/acre	Waste Type	Canister Orientation
Salt	60	SF-PWR	Vertical
	30	SF-PWR	Vertical
	60	SF-PWR	Horizontal
	30	SF-PWR	Horizontal
	150	HLW	Vertical
	75	HLW	Vertical
	150	HLW	Horizontal
	75	HLW	Horizontal
Basalt	190	SF-PWR	Vertical
	95	SF-PWR	Vertical
	190	SF-PWR	Horizontal
	95	SF-PWR	Horizontal
	190	HLW	Vertical
	95	HLW	Vertical
	190	HLW	Horizontal
	95	HLW	Horizontal
Granite	190	SF-PWR	Vertical
	95	SF-PWR	Vertical
	190	SF-PWR	Horizontal
	95	SF-PWR	Horizontal
	190	HLW	Vertical
	95	HLW	Vertical
	190	HLW	Horizontal
	95	HLW	Horizontal
Shale	130	SF-PWR	Vertical
	65	SF-PWR	Vertical
	130	SF-PWR	Horizontal
	65	SF-PWR	Horizontal
	130	HLW	Vertical
	65	HLW	Vertical
	130	HLW	Horizontal
	65	HLW	Horizontal

GEOLOGIC REPOSITORY CONSEQUENCE ANALYSIS

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A methodology has been developed for obtaining an initial perspective on the potential radiologic-dose consequences from deep geologic repositories that would result from a scenario involving groundwater penetration into a repository, leaching of the radionuclides from the waste, and transport of the nuclides to the biosphere where they may interact with humans.[1,2] This methodology is now being applied in two NWTS/ONWI programs:

(1) A Systems Study on Engineered Barriers, which is designed to develop a perspective on what benefit may be obtained by reduced leach rates, more durable containers, or increased sorption of nuclide ions. This benefit is measured in terms of incremental dose reduction due to improved barriers.

(2) An Ore Bodies Comparison Program, which involves applying the same methodology to naturally occurring uranium ore deposits, in order to develop a perspective on the potential dose consequences from ore deposits relative to those from a man-made repository.

Systems Study on Engineered Barriers

The development and characterization of waste forms, containers, and other engineered barriers destined for use in the isolation of nuclear waste in deep geologic repositories has progressed to the point where there are several options and combinations of barrier systems that are available to help assure safe disposal of nuclear wastes. However, a rigorous basis has not yet been developed to define whether various concepts or products are required or desirable, or how durable they should be for how long. This analysis is an attempt to develop an initial perspective on what is needed in order to establish that basis. The intent of the analysis is to roughly determine what incentives exist for providing highly durable engineered barriers for the isolation of radioactive waste in a deep geologic repository, based on what effect, if any, the engineered barrier may have on the ultimate potential dose to humans.

The highest risk path from the storage site through the geosphere to the biosphere is assumed to be the "leach incident pathway". The geologic isolation system provides barriers to the process by two general means: (1) containment of the waste for a length of time sufficient to allow the hazardous nuclides to decay to innocuous levels such that unrestricted release to the environment presents no radiological hazard, and (2) limiting the rate of nuclide release to the biosphere so that nuclide concentrations in the constantly renewed local surface water and air never exceed acceptable limits.

The actual repository system will provide protection by using some effective balance of these two means. The available system components that provide these means for protection can be reduced to: (1) the waste form, (2) the containment package, and (3) the geology (including hydrology).

The radionuclide inventory used in the analysis is one-fifth of the total for the spent, unreprocessed fuel (no-recycle case) accumulated in the U.S. through the year 2050 that was used in the DEIS.[3] The assumption was that there would be five operating repositories, each containing an equal fraction of the total U.S. spent-fuel accumulation.

The analysis is being accomplished with the aid of a combined geosphere-transport[4] and biosphere-transport/dose[5-7] computer code. The geosphere-transport model simulates the release of radionuclides from a repository through contact and leaching by flowing groundwater, subsequent transport of the nuclides in the groundwater (including the effects of convection, sorption, and radioactive decay), and the ultimate release of the nuclides to the surface water (lake or river). In order to simplify the analysis, numerous parameters were reduced to terms representing the net effect of several more complex phenomena. Output data of the geosphere model are the input data for the biosphere model. The biosphere model predicts the radiation dose to humans from the nuclide release to surface waters via several pathways, including direct ingestion of water, aquatic foods, and irrigated food products, as well as irradiation from shoreline deposition and swimming. Food-chain reconcentration/accumulation effects are included.

The analysis is being applied to reference repositories in salt and basalt reference geologies this year, and will be applied to reference granite and shale sites later. The results of this analysis, combined with in situ performance predictions for various engineered barrier systems, lay the groundwork for a cost/benefit analysis of the geologic repository/barrier system.

Ore Bodies Comparison

It has been suggested that one general criterion for deep geologic repositories containing nuclear waste is that they should result in no greater radiological risk than that due to naturally occurring uranium ore deposits.

Descriptive data for two major ore deposits were gathered from available literature and discussions with experts at USGS in Grand Junction, Colorado. Potential dose consequences from these ore deposits were then calculated using the same methodology as was applied in the analyses of nuclear waste repositories in order to compare the relative potential consequences. As a check on the ore bodies release estimates, the calculated results from the ore bodies analysis have been compared to actual measured surface-water concentrations of radioactive material that apparently had their origin in natural ore bodies.[8]

Initial results of this work indicate that the potential radiological consequences from a waste repository containing spent nuclear fuel and those from a large uranium ore deposit are quite comparable. Recent improvements in the biosphere transport model and extensive data on measured uranium and radium concentrations in U.S. surface waters are being incorporated into the analysis to refine the comparison.

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PERFORMANCE CHARACTERISTICS AND COSTS OF SELECTED ENGINEERED-BARRIER CONCEPTS

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Science Applications, Inc.

This study is concerned with the disposal of unreprocessed BWR and PWR fuel elements in salt, shale, basalt, or granite repositories. Use of a system of engineered barriers as containment in addition to the geologic media is being considered. The objectives of the work are to develop a scoping model of barrier performance, apply the model to a representative spectrum of designs, and perform estimates of incremental cost for development, manufacture, and emplacement relative to a baseline minimal package. "Performance" in this study is related only to ultimate population dose after repository closure and not to other factors such as transportation. Conceptual designs studied were described with retrievability in mind, although this was not the focus of this study. This work is a part of the Systems Study on Engineered Barriers. The other portions of the systems study were discussed previously by M. Cloninger. Results of the incentives study are used to relate performance and cost to a benefit in terms of risk to the population. The purpose of the overall effort is to provide a tool for screening candidate concepts and guiding R&D programs for barrier concepts.

Barrier performance is determined in terms of two parameters: time of initial release to the geosphere and duration of the release (leach time). Time is measured from a scenario zero time when the repository has been sealed and is assumed saturated with water. The performance is related to the incentives study, which evaluates population dose in terms of a three-dimensional surface: time of release versus duration of release versus transit time through the geosphere.

The performance model treats a barrier package as a series of layers each consisting of a filler and a solid wall (in some cases, no filler is present). For each time interval, temperature and pressure are calculated to determine stresses on the barrier element. If filler is present, a filler compaction due to geologic-medium creep strain is calculated. The minimum wall thickness to withstand the resultant stress is calculated. Corrosion attack (effective wall thinning) is accounted for in each time interval. Eventually, the layer fails. This is defined in the model as that time when the wall no longer meets ASME code (Section VIII) for the net stress calculated. Once failed, the layer is removed from consideration, and water is assumed to contact the next layer. The process is repeated until water penetrates the inner barrier and reaches the fuel. Then a leaching model is applied along with an adsorption model (if a migration barrier is present) to determine release time and duration of release.

The fuel material is the "once-through fuel cycle" material for PWR and BWR waste referred to in the draft GEIS.[1] Basic information on the reference repositories is given in Table 1. These are consistent with the GEIS reference repositories, as are the rock properties. Materials and designs for barrier packages were chosen to give a range of cost and performance. Table 2 is a matrix of basic concepts studied. Table 3 is a listing of representative materials studied. While numerous materials can be chosen as alternatives, many are similar in cost and/or performance. Many other materials will be considered in future barrier design studies and development work. The key concern was to identify where additional barrier cost yields little increased benefit. A zero barrier baseline package (see Table 2) was defined as a basis for all relative costs and performance. An example of a highly complex PWR package for a salt repository is described in Table 4.

Preliminary results indicate some controlling factors in a barrier performance. In media which display significant creep strain (such as salt and, possibly, shale) a high crushing force builds up on the package. Wall thicknesses required to withstand such forces are large. In comparison to the compressive force applied to the package by the surrounding semiplastic rock, corrosion is not an appreciable effect in determining package cost or performance. However, loosely packed backfill and filler materials can provide long-term absorption of creep movement, preventing rapid pressure buildup due to rock creep. In granite or basalt, such crushing is negligible, because of near-zero creep rates. Thus, considerable expense in package metal is incurred for implacement in high-creep-strain media, while relatively thin containers may last for thousands of years in noncreeping media. Furthermore, the use of an overpack in creeping media does not appear warranted, as a thick sleeve is a better alternative. Once the sleeve collapses, the overpack and canister will not last any significant time. Conversely, a corrosion-resistant overpack can increase performance in basalt or granite, but a thicker canister will also. The aggressive corrosion behavior of brine at high pressures and temperatures (especially oxygenated brine) requires significant additions of metal for corrosion allowance, but this is also overshadowed by the problem of external crushing forces. In the low-ionic-strength waters associated with non-salt media, thin-walled barrier elements last an indefinite time; the period is difficult to predict because of the enormous corrosion-data extrapolation in time required.

While migration barriers can be effective (on the order of 100 to 1000-year delay in release), they are usually not a significant improvement over the excellent migration-barrier characteristics of the geologic medium itself. However, when a backfill is employed as a cushion to absorb rock-creep strain, its ion-exchange capability would serve double duty as an aid in limiting radioactive isotope migration.

Packages for protection of spent fuel elements emplaced in geologic media can be developed that provide sufficient delay before release of radioactivity to the point that additional delay would not result in a further reduction of risk.

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TABLE 1. SUMMARY OF REPOSITORY AND SITE PARAMETERS

Host Medium	Repository Depth Meters	Repository Depth Ft	Groundwater[a]	Creep (cm/yr)	Emplacement Density, Canister/ha PWR	Emplacement Density, Canister/ha BWR
Salt	580	1900	WIPP Brine B	0.64	140	450
Basalt	600	1970	Low ionic strength	0	345	1125
Granite	620	2035	Low ionic strength	0	345	1125
Shale	460	1510	Low ionic strength	0.25	225	500

[a] Both oxic and anoxic conditions considered.

TABLE 2. MATRIX OF BARRIER CONCEPTS STUDIED (PRELIMINARY)

Barrier Package	Stabilizer	Canister	Overpack	Sleeve	Backfill	Number of Barriers
Concept A	Gas and solid	Corrosion-resistant	—	—	Special	2
Concept B	Gas and solid	Mild steel	Corrosion-resistant	—	Special	2
Concept B1	Gas and solid	Mild steel	—	Corrosion-resistant	Special	2
Concept C	Gas and solid	Corrosion-resistant	Corrosion-resistant	—	Special	3
Concept C1	Gas and solid	Corrosion-resistant	—	Corrosion-resistant	Special	3
Concept D	Gas and solid	Mild Steel	—	Corrosion-resistant with coating	Special	3
Concept D1	Gas and solid	Corrosion-resistant	—	Corrosion-resistant with coating	Special	4
Concept D2	Gas and solid	Corrosion-resistant	Corrosion-resistant	Corrosion-resistant with coating	Special	5
Baseline	Gas	Mild steel	—	—	—	0

TABLE 3. REPRESENTATIVE MATERIALS FOR BARRIER PACKAGE COMPONENTS (PRELIMINARY)

Component	Salt Host Medium	Non-Salt Host Medium
Stabilizer	Helium Al_2O_3 blocks Metal blocks[a]	(same)
Canister	Mild steel Zircaloy Inconel	Mild steel Zircaloy 304 stainless steel Inconel
Overpack	Lead Zircaloy Copper Inconel	Lead Zircaloy Thick cast iron Copper Inconel 304 stainless steel
Sleeve	Inconel Zircaloy Cast iron (with coating) Copper Mild steel (with coating) Mild steel	Inconel 304 stainless steel Zircaloy Cast iron (with coating) Copper Mild steel (with coating) Mild steel
Coatings	Polymers Fired coatings	(same)
Backfill	Clay Zeolites Al_2O_3	(same)

(a) To be same as canister material.

TABLE 4. EXTREME CASE: A MULTIELEMENT SALT-MEDIUM PACKAGE

Barrier Element	Inside Diameter, cm	Outside Diameter, cm	Material	Function
Stabilizer	30.5	34.3	Inconel	Heat transfer and geometric retention
Canister	34.3	35.6	Inconel	Water barrier
Filler	35.6	39.4	Fine sand	Cushion
Overpack	39.4	40.6	Inconel	Water barrier
Filler	40.6	43.2	Fine sand	Cushion
Sleeve	43.2	58.4	Cast iron	Water barrier
Backfill	58.4	122	Sand + 10% bentonite	Cushion and migration barrier

AN OVERVIEW OF THE ECONOMICS OF NATIONAL WASTE TERMINAL STORAGE

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Introduction

The objective of this NWTS/ONWI study is to illustrate cost and economic effects of program alternatives. To meet this objective, cost trends are analyzed, economic methods are evaluated, and economic and technical data are compared on consistent bases. In general, the economics program obtains cost data from published and unpublished studies, such as conceptual engineering-design studies, as well as directly from those working on waste-management projects, for example, the Savannah River Laboratory. When data from several sources are used in the economic analysis, they are adjusted to provide a consistent basis and a common set of assumptions. Source data and any necessary adjustments are carefully documented to provide traceability.

The scope of the economic analyses through FY 1979[1-3] has included studies of the government's planned cost-recovery methods for management of commercial spent nuclear fuel and studies of repository expected cost trends. This paper provides a brief discussion of each.

Spent-Fuel Studies

Current government policy is to accept spent fuel from utilities in return for a fee which will fully recover all relevant government expenses. This policy is implemented specifically by calculating the charge ($/kg) which equates the projected present values of government costs and revenues. No credit is given for uranium or plutonium in the spent fuel. These studies compare alternative calculations and identify the most sensitive parameters.

Repository cost analyses have included comparisons of recent engineering-design studies and preparations of summaries that can provide a basis for communicating these results. This work has included uncertainty analyses and repository parametric analyses. Uncertainty analyses have researched the effects of variations that may be caused by cost-component uncertainity, design/scope uncertainty, and economic uncertainty. A parametric study of repository costs is planned covering 40 cases in which the effects of 13 of the most sensitive variables will be analyzed.

This program has maintained close programmatic interrelationships with the responsible government offices for each of the cost centers included in the government-fee calculation. These activities have been concerned with away-from-reactor (AFR) storage, transportation, research and development (R&D), and repository-cost estimating activities for all geologic media.

Four methods of calculating a fee for government spent-fuel storage and disposal services are compared in Table 1. The annual average method estimates the costs of an average year of operation, including a fixed capital charge for constructed facilities, discounts this cost over the life of the facility, and divides the result by the discounted receipts. This method has the advantage of simplicity; however, it does not illustrate secondary effects, such as phase-in periods in repository receipts.

The venture method treats one repository as a single project. Thus, costs are discounted over one repository's discounted total recepts to estimate a fee. This method eliminates ending-value financial assumptions, but leads to difficulty in maintaining continuity in price recalculation through one repository filling and a second coming on-line.

The campaign methodology requires the estimation of all costs required in a given time (the campaign) and equating present values of costs and revenues for the receipts in that time frame. Pricing of government uranium-enrichment services uses this method. Recalculation is simplified by dropping the past year and adding a future year at each annual recalculation. This method requires depreciation assumptions to account for the value of existing facilities at the end of the campaign period.

The use-based method first generates a cost for each cost center. These costs are allocated between storage (AFR) and disposal (repository) users, and disposal-only users, according to the applicable discounted receipts. This results in cost-center prices for each type of user. The use-based method then adds all cost-centers component prices for storage and disposal and for disposal only to arrive at separate fees for the two types of users. This method was used in the first DOE charge calculation[4] and its supporting documentation[6]. Its advantages include recovery of costs for each component from only users of that component, and the advantages of the campaign method since a constant time period is used. This method raises the price charged to utilities which must use government AFR storage, but reduces the charge to utilities of use-only disposal.

Figure 1 illustrates the resulting prices of 45 cases using the venture and campaign methods.[1] The most sensitive parameter in this study was repository capacity, which varied from 44,200 MTU to 104,700 MTU as a function of thermal criteria based on retrievability requirements. Figure 1 shows that almost all prices fell in two bands depending upon retrievability. Figure 2 shows the breakdown of all undiscounted costs for the case with 5-year retrievability and 5-year-old transfers (Case F8) including the cost of transportation from the nuclear power plant. Transportation is the largest cost. Transportation costs included in the government fee are about one-sixth of total transportation costs. Note also that geologic-repository costs are dominated by the operations category.

Figure 3 depicts the national average price that results from the use-based method. These results were calculated by weighting the separate fees for storage and disposal, and disposal only [4], by the amount of fuel that would pay the fee. A band of prices is again observed, with two exceptions for the low-demand and late-repository cases. These results appear to give different sensitivities to parameters such as repository capacity and repository delay. Careful analysis reveals the underlying factor to be the quantity of spent fuel over which the costs are distributed. In venture cases, the quantity of spent fuel over which the costs are distributed is determined by repository capacity, whereas in campaign cases the quantity of spent fuel is determined by demand and the time period of the campaign. AFR storage-requirement sensitivity is greater using the use-based method, principally because its effects occupy a greater percentage of the time period since time is not extended, as it is in the venture method.

Repository Cost Analysis

An overview of recent engineering studies[6–9] is summarized in Table 2. Studies of the cost-category definitions will be published soon which will describe the differences in coverage and the taxonomy which exist between these studies. As expected in independent studies, no two are exactly comparable.

Cost-component uncertainty studies have resulted in the distributions shown in Figure 4. These distributions assume cost components follow the Maxwell distribution which was selected from four candidate distributions after study of cost differences reported by the General Accounting Office[10] and Rand Corporation[11]. These distributions represent the minimum which must be expected at this time in total project cost at completion.

Economic-parameter uncertainty studies have recently indicated that non-linear effects of parameter variations can lead to uncertainties exceeding 10 percent of the inflated total cost of the project at completion. This finding assumes inflation is modeled as a constant base rate plus a random component symmetrically distributed with zero mean and standard deviation one-half the base rate. High values in this project are the result of long project life and high operations costs relative to construction.

Future Activities

In the future, economic-analysis efforts will be directed to updating the estimate of the government fee, and continuing efforts in uncertainty and parametric analyses. The next fee calculation will address the implications of the recommendations of the recent Interagency Review Group.[12] These recommendations are expected to increase AFR storage requirements, incorporate multi-media research and development costs, and address regional repository construction. Scope/design parameters are expected to represent the greatest uncertainty when compared to cost-component and economic uncertainties. Parametric analyses are expected to result in a ranking of sensitivity of NWTS costs to the input parameters.

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TABLE 1. COMPARISON OF NWTS ECONOMIC METHODOLOGIES

Methodology for Computing Government Fees for Spent-Fuel Storage and Disposal Services				
	Annual Average	**Venture**	**Campaign**	**Use-Based**
DESCRIPTION	Estimate the average year	One repository	Defined time period	Assign cost by requirement
ADVANTAGES	No funding spreads required	Minimizes financial assumptions	Continuous, enrichment precedent	Recovers costs from users by component campaign advantages
DISADVANTAGES	Does not illustrate secondary effects	Difficult to maintain continuity	Requires financial assumptions	Requires a separate charge for each type of user

TABLE 2. SUMMARY OF RECENT STUDIES

	Repository Study[a]				
	NWTSR1[6]	NWTSR2[6]	GEIS[7]	SFDS[8]	TM-36[9]
Corporate Author	Stearns-Rogers Engineering Co.	Kaiser Engineers, Inc.	DOE et al	Bechtel National, Inc.	SAI/PBQD
Media	Domed salt	Bedded salt	Salt, basalt, shale, granite	Generic salt	Salt, basalt shale, granite
Waste Form	HLW & spent fuel	Spent fuel	HLW & spent fuel	Reference HLW & spent fuel	HLW & spent fuel
Location	Gulf Coast	Midcentral Plains	Generic	Generic	Generic
Thermal Loading (kw/acre)	150-40	60-40	125-40	150-60	190-36
Number of Shafts	5-6	4-5	5	5	5-7
Foundation	40-ft. caissons	Slab	Slab	Slab	Slab
Air Conditioning	Yes	No	No	No	No
Rock Disposal	Off-site	On-site	On-site	On-Site	On-site
Cost Base Year	January, 1978	January, 1978	1978	January, 1979	1977
Total Project Cost Range ($B)	1 8-2.5	1.3-1.6	1.6-5.5	2.7-4.7	1.9-5.7

[a] Superscript number refer to documents cited under "References".

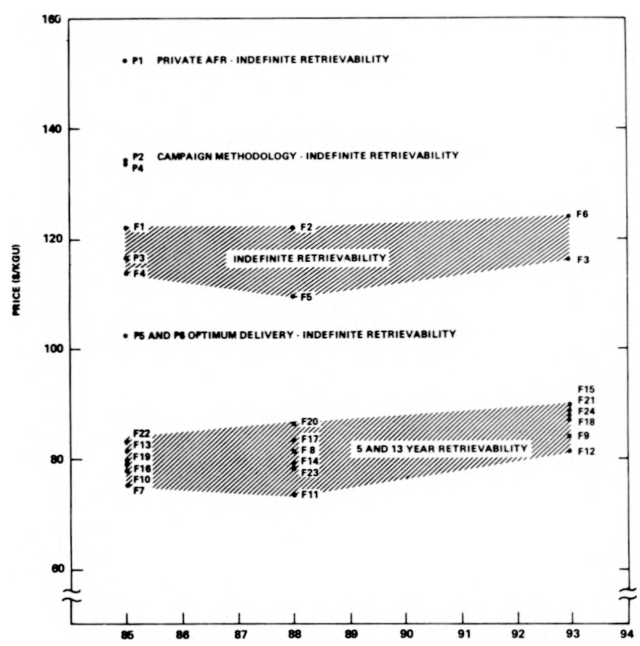

FIGURE 1. VENTURE AND CAMPAIGN METHODOLOGY GOVERNMENT FEE RESULTS

See Reference (1) of coding.

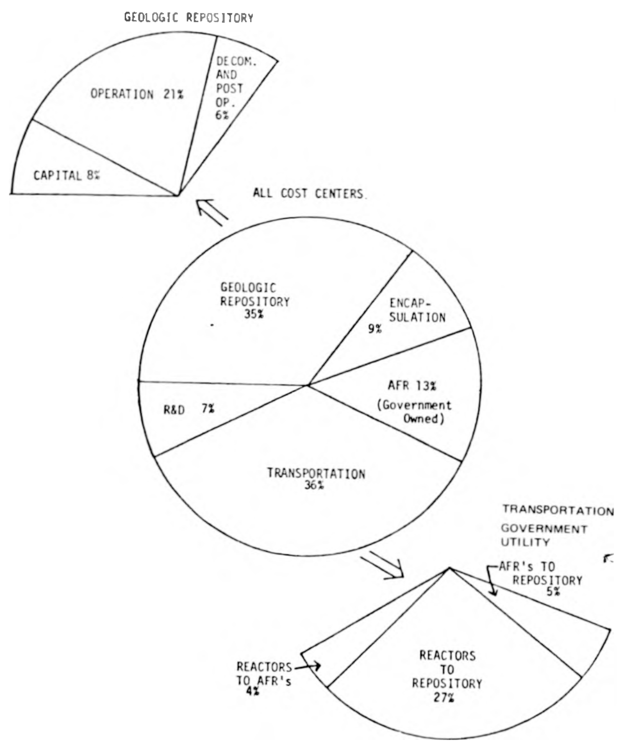

FIGURE 2. SPENT-FUEL STORAGE AND DISPOSAL-COST RESULTS FOR 5-YEAR RETRIEVABILITY OF 5-YEAR-OLD FUEL TRANSFERS

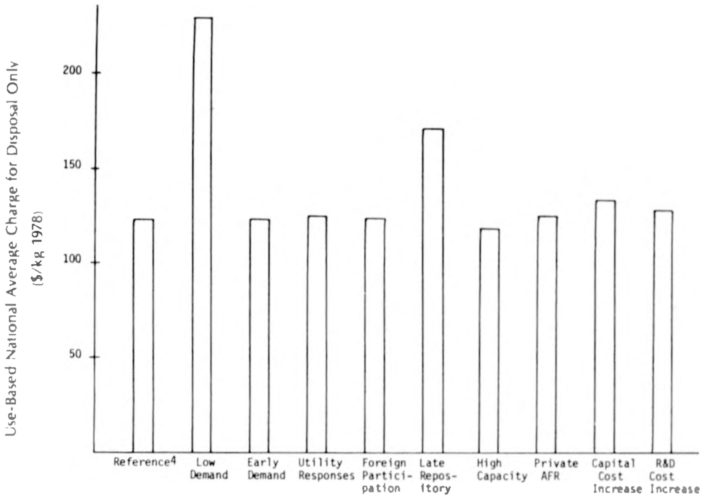

FIGURE 3. USE-BASE WEIGHTED AVERAGE CHARGE FOR STORAGE AND/OR DISPOSAL METHODOLOGY GOVERNMENT-FEE RESULTS

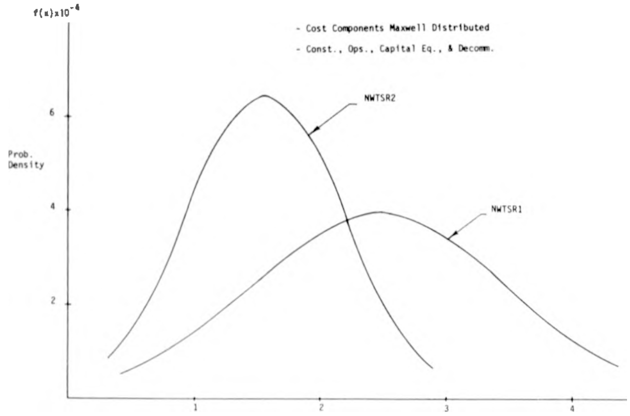

FIGURE 4. TOTAL NWTS PROGRAM COST COMPONENT UNCERTAINTY

DEVELOPING A SAFEGUARDS PROGRAM FOR THE NUCLEAR WASTE ISOLATION SYSTEM

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The objective of the Allied-General Nuclear Services (AGNS) program is to develop safeguards and security criteria for the Waste Isolation System (WIS). The recent AGNS direct experience in developing criteria and specifications for a major nuclear fuel-cycle facility and directing the engineering, licensing, and construction of such a facility will be applied to the challenging task of developing criteria that are not only adequate but cost-effective.

The criteria under development by AGNS will be based on current, proposed, and projected regulatory requirements and AGNS's experience in developing a fully integrated and coordinated advanced safeguards system. The work during 1979 has included: (1) preparation of a scoping study, (2) development of a safeguards program plan, and (3) technical review of documents related to the NWTS Program.

The criteria will be used to design safeguards and security systems for the various major elements of the WIS. Since the safeguards and security systems will substantially affect the design of the facilities, it is essential that the criteria be developed on a schedule consistent with, and adequate for, initiation of Title I design efforts. In addition, the performance/design criteria developed under these efforts will be suitable for use by scientists and engineers for licensing and planning for operation of the WIS.

The scoping study, which will be completed early in 1980, is the major task in the 1979 AGNS program. First, a comprehensive literature search was performed to identify existing safeguards and security technology and regulations applicable to the WIS. From this search, an extensive bibliography was prepared. Each entry is being annotated as to content, and then an assessment is being made of the applicability of the safeguards technology. Secondly, an effort is being made to formulate a "consensus" view of future safeguards trends.

Using this information, a scoping-study report is being written. This report will identify the major elements of the WIS and the waste materials to be protected, and present the AGNS assessment of the perceived threat levels. It will then provide definition of the basic safeguards and security criteria and requirements for the WIS. These criteria will cover such areas as materials control, materials accounting, access control, intrusion detection and alarm assessment, containment and surveillance, and contingency planning.

In addition, the scoping-study report will contain preliminary performance/design criteria for the safeguards and security system for a geologic repository. This work is focusing on the needs revealed in the review of the Preliminary Information Report/Conceptual Repository Reference Design (PIR/CRRD) documentation, which encompasses emplacement of spent LWR fuel in a geologic repository.

The program plan, which also will be completed early in 1980, will include the work breakdown structure, milestone and cost schedules, and logic networks for the ongoing effort. The work will focus on preparation of detailed design/performance criteria, design specifications, and procurement specifications for the safeguards and security systems for the geologic repository. In addition, design/performance criteria will be established for safeguards and security systems for other elements of the WIS.

AGNS has also made technical reviews of the safeguards and security sections of a number of reports related directly or indirectly to the NWTS/ONWI Program. AGNS's reviews and the viewpoints and comments offered were based on AGNS's background knowledge and experience in dealing with architect/engineers and system/equipment designers during the design, construction, and licensing of the Barnwell Nuclear Fuel Plant (BNFP).

SESSION IIIB

TECHNICAL STUDIES IN THE NWTS/ONWI SITE AND REPOSITORY LICENSING PROGRAM

OVERVIEW OF SITE-QUALIFICATION AND LICENSING ACTIVITIES

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Site-qualification and licensing activities are directed at providing the Department of Energy with technically qualified sites for waste isolation and with the programs and documentation required to obtain Nuclear Regulatory Commission licenses to construct and operate repositories at selected sites. The site-qualification process starts with the identification of relatively large geographic areas which, based on information available in the technical literature, are considered to have a strong potential of containing hydrogeologic systems suitable for waste isolation. Through successive phases of geologic, hydrologic, and environmental studies, additional information is gained which indicates those portions of the areas under study showing the strongest potential for containing suitable waste repository sites. Less suitable areas are screened out and more suitable areas are studied in greater detail. Through the successive screenings, the areas under study are reduced from regions covering tens of thousands of square miles and several states, to areas of approximately 1000 square miles, and then to locations of approximately 30 square miles. Criteria have been developed to evaluate the information developed in the siting studies and to form a basis for the screening process.

At the present time, the NWTS/ONWI program siting studies are actively under way in salt formations in Louisiana, Mississippi, Texas, and Utah. Studies are currently being planned in nonsalt formations; however, the specific areas which will be studied have not yet been determined.

The sites which emerge from the screening process with the strongest potential will be evaluated from a safety and environmental point of view using available models. Those evaluations will be documented in individual Site-Qualification Reports. The Site-Qualification Report evaluations will not have the same degree of sophistication and detail as will be contained in the Safety Analysis Reposts prepared for licensing. The Site-Qualification Report evaluations will be directed at determining whether there is reasonable confidence, based on the information available, that further study will provide the high degree of confidence in the site that would be required for licensing.

At some point (dependent upon the President's waste management policy) a site will be selected for further study, leading to NRC licensing. It is likely that the selection process will require heavy state involvement and will be conducted in accordance with the NEPA process. The selection process will also involve land acquisition proceedings and notification of the NRC by the DOE of the pertinent characteristics of the site selected.

The notification of NRC will be under the heading of an "early site review". The vehicle for providing that notification will be the Site-Qualification Report for the site selected. Following site selection, additional geologic, hydrologic, and environmental studies will be conducted and detailed site-specific design activities will commence in order to provide the information required to develop a Safety Assessment Report and Environmental Report for licensing. At the present time, ONWI is developing SAR and ER format and content guides along with plans and programs to assure the availability of the technology and data required to implement the guides.

More detailed information on the site qualification and licensing elements of the NWTS/ONWI program will be provided in subsequent discussions.

A PLAN FOR LICENSING GEOLOGIC REPOSITORIES

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The Energy Reorganization Act (ERA) of 1974 gave DOE (then ERDA) the responsibility for developing and operating facilities for the long-term isolation of high-level radioactive wastes. The NRC was given regulatory responsibility for licensing such facilities. The responsibilities delineated by the ERA placed both DOE and NRC in positions which were new to them with regard to licensing. Although NRC instituted programs to develop the technology and standards required to regulate high-level waste disposal, little information has been made available by NRC to guide DOE in preparing for licensing. The enormous wealth of information developed for reactor licensing is of limited value to repository licensing because the technical objectives, evaluation techniques, and safety and environmental concerns are so vastly different. In order to bridge the gap until definitive information is promulgated by the NRC, NWTS/ONWI is developing criteria, format guides, and evaluation techniques which are believed to be consistent with those which will be required by the NRC when its requirements are solidified.

The basis for predicting the general thrust of future requirements by NRC includes reviews of preliminary draft information available to the public from the NRC staff, proposed policies and criteria by NRC and EPA, study results published by consultants to NRC and EPA (e.g., National Academy of Science studies), inferences drawn from those elements of repository licensing which have parallels in NRC facility or materials licensing, and information exchange meetings with NRC.

To date ONWI has prepared format and content guides for a Safety Analysis Report and an Environmental Report; strategies which ONWI believes DOE should adopt to prepare for the licensing process and avoid licensing delays which would not contribute to the safety or environmental acceptability of a repository; and plans for implementing those strategies. All of those efforts are being coordinated with the other groups in the NWTS program in order to assure consistency in approach.

A significant part of the preparation for licensing is the preparation of the Preliminary Information Report (PIR). The PIR is a combined Preliminary Safety Assessment Report and an Environmentl Report, prepared for a fictitious repository site, which is a composite of information from several geographic locations, none of which could be used for an actual repository due to prior use conflicts. Internally, the value of the PIR is that it will (1) indicate the suitability of the present programs to produce high-confidence licensing documents, (2) indicate areas where additional emphasis is needed, (3) highlight issues which require resolution, (4) help identify the actions required to resolve such issues, and (5) provide insight into preparing detailed schedules and delineating responsibilities for developing an actual PSAR and ER at a later date. If the PIR is reviewed in detail by NRC, it will provide NRC with similar insights into its own program and aid in identifying the real issues of concern for licensing. Both of these benefits could reduce the potential for unnecessary time losses in licensing.

Although many schedules have been prepared by NWTS program participants for licensing, such schedules have value only as planning aids to integrate various facets of the programs. It will not be until a national policy and schedule for repository development has been approved by the President and the NRC has solidified its own requirements for licensing repositories that definitive schedules can be developed and pursued with confidence.

A PLAN FOR QUALIFICATION AND SELECTION OF GEOLOGICAL REPOSITORY SITES

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This paper presents a discussion of the processes for the qualification and selection of geologic repository sites. The site-qualification process will yield techniclly qualified site; the site-selection process will choose, from among technically qualified site, a site (or sites as required) for subsequent, detailed, site-specific designs, assessments, and analyses and other steps intended to culminate in a construction permit. These processes are in the planning stage; this discussion therefore presents concepts which may be revised.

Background

The National Environmental Policy Act of 1969 (NEPA) requires [in Section 102 (c) (iii)] for "major Federal action significantly affecting the environment" that "alternatives to the proposed action" be considered. Among the kinds of alternatives which must be considered are alternative sites. The considerations of alternatives should [from Section 101 (b) (3)] . . . "attain the widest range of beneficial uses of the environment without degradation, risk to health or safety or other undesirable and unintended consequences". In partial conformance with these requirements the site-qualification process will identify candidate sites and determine whether or not each candidate meets certain site-qualification requirements. The selection process will choose sites for construction-permit applications from qualified site options on national need, institutional, and other applicable bases.

National policy with regard to nuclear waste management is in development; the present status of the policy is in the form of recommendations of the Interagency Review Group (IRG) which are presented in their final report (TID-29442, March, 1979). Several aspects of recommended national policy impose requirements on the site-qualification and selection processes; other issues for which policy is undecided impose the need for processes which can adapt to ranges of policy options. The policy with regard to alternative considerations is not decided; e.g., IRG Strategy I or II would require, for the first repositories, a consideration of candidate sites in rock salt (or in other media for which the information is available at the times of selection), but Strategy III would require the consideration of sites in three to five geologic media. The siting processes must therefore be adaptable to each of these strategies. The IRG favored regional repositories; hence the processes must be able to deal with that possible national policy. The IRG recommended "state and local consultation and concurrence" and a State Planning Council to "provide state perspectives for the development of the National Waste Management Plan, the site characterization programs...". and "assist DOE and the states in recommending proposed sites...". IRG also stressed the importance of "public participation" and the goal of a "social consensus". The siting processes therefore must incorporate mechanisms for effective state and local consultations and public participation.

The DOE has published a generic DEIS on "Management of Commercially Generated Radioactive Waste" (DOE/EIS-0046-D, April, 1979) in which (on pp. 3.1.17 et seq.) the site-qualification process for Conventional Geologic Disposal is discussed. The GEIS presents a "purely technical" approach to site qualification which emphasizes the special importance of earth-science considerations. It also points out the importance of "balancing short- and long-term risks" and of "socioeconomic impacts". The qualification/selection processes should therefore emphasize the earth-science considerations but also include other aspects.

Site-Qualification Process

The site-qualification process will be designed to produce the documentation (of data, assessments, analyses, reviews, consultations, concurrences, and public participation) to defend the winnowing down of the total search region to candidate sites, and to determine technical qualification of candidate sites. This process will be based on certain procedures; the procedures will be flexible enough to provide guidance to all the NWTS site searches, but will be specific enough to ensure that the product of each site-search program will be comparably qualified.

The winnowing-down procedure will take the general form outlined in the GEIS and will employ the NWTS Site-Qualification Criteria [ONWI-33(2), Draft July 30, 1979], but the details of the procedure will differ among programs in accordance with the total search area (i.e., the contiguous 48 states, the Nevada Test Site, etc.) and with the search approach (i.e., the host-rock approach or systems approach). The data development procedures to support the winnowing down will differ among programs, but the data bases for candidate sites should allow qualification against generic site requirements.

The decision steps in the winnowing down should be documented with reports which summarize the technical (and other) arguments and state the decision. The final decision documents in the site-qualification process should be a Site-Qualification Report for each candidate site which presents a summary of the supporting information and the conclusion that the candidate site is either qualified or disqualified.

General procedures for state and local consultations and public participation are expected to be included in the plan to assure a consistency in the level of the efforts. Detailed procedures will vary in accordance with the needs and desires in each state. Procedures should also include mechanisms for reviews at each decision step. Procedures for consultations, public participations, and reviews should include mechanisms for documenting comments and program responses.

Site-Selection Process

The site-selection process will accept qualified sites as input and will output a site (or sites, as required) with the rationale, concurrences, and approvals which may be required for the performance of detailed, site-specific designs, assessments, and analyses, submittal of an ER and PSAR, and other steps to acquire a construction permit.

The site-selection process must be capable of serving a range of possible national policies; e.g., the process may be used to select one salt-dome site from among a number of salt-dome sites (as in IRG Strategy I); it may be used to select a site for a regional repository from among candidate sites for repositories in a variety of geologic media (as in IRG Strategy III for a regional repository). Indeed, other variations of the site-selection mission could evolve; therefore the site-selection process will be planned in terms of flexible procedures. It may be useful to employ a banking concept in which qualified sites are put into a bank for subsequent site selection.

Procedures will be needed for land withdrawal; e.g., use of lands managed by the Bureau of Land Management (BLM) will require appropriate application and a decision process by BLM. A procedure for early site review by the USNRC may be helpful and/or required. State interactions at the site-selection stage may include a concurrence process with a State Planning Council, such as recommended by the IRG, or comparable other arrangements as may develop from Congressional action. In any case, the site-selection process will include procedures to assure state and local consultation and concurrence and public participation. Other reviews such as peer reviews and reviews by other agencies and institutions are being planned.

It is planned that a technical report will be produced at this stage — a Site-Recommendation Report — in which the pertinent qualified sites will be compared in terms of relative risks, impacts, and costs projected for a repository at each site. The report would fulfill the NEPA documentation associated with site selection and might also serve other needs (i.e., early site review by NRC). The site-selection process should also include guidance relative to scheduling of necessary steps in the licensing process.

PLANS FOR CONDUCTING ENVIRONMENTAL SURVEYS

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The site-qualification process requires a means of winnowing down the area of the contiguous 48 states to candidate sites and of qualifying candidate sites against site specifications. In the cases of NWTS/ONWI site searches, the acquisition of the data needed to support these qualification efforts involves two major activities: geologic surveys and environmental surveys. This paper describes the planning that has been done and is projected to be done with regard to the environmental surveys.

Geologic studies are conducted to identify areas having the geologic/hydrologic system characteristics required for safety and engineering design of a deep geologic repository. Environmental studies, conducted in parallel with the geologic evaluations, are necessary to round out the evaluation process and assure that the comprehensive information necessary to meet the requirements of the site-qualification process and of the National Environmental Policy Act will be available. The first step in the environmental-assessment process is the development of plans for the performance of each of the environmental studies, i.e., for regions, areas, and locations. The plans are circulated for review and comment. Reviewers include state legislators, state agency personnel, regional experts in the environmental disciplines, individuals representing environmental coalitions, and interested citizens. This external review process provides a means of both consultation and concurrence and public participation relative to the planning and execution of environmental surveys.

Preliminary environmental studies are conducted to characterize an identified region. Using available data from the literature, the region is screened for promising areas for further study based on an assessment of potential environmental impacts or risks that could result if a repository were to be located in that region. The parameters employed cover all aspects of the total environment, including geography, terrestrial and aquatic ecology, hydrology, meteorology, land and water resources, and land use and demography, as well as economic, historic, institutional, and societal factors. The results of the regional studies are circulated widely for review and comment.

The second phase of the environmental studies is conducted to characterize each identified study area in terms of those parameters that are required to screen the area for promising locations for further study and to evaluate potential impacts or risks of a waste repository if it were to be located in that study area. The environmental characterization of study areas may require acquisition of data not available in the existing literature. Again, area-characterization reports will be circulated widely for review and comment.

The third phase of the environmental studies will characterize locations in detail; characterizing those parameters that are necessary to assess potential repository impacts for (1) site qualification and (2) licensing documents. Those studies will require the collection of field data in addition to those data presently available. Study plan documents for all three study phases have been prepared. These planning documents will be followed by program contractors in performing the indicated environmental data gathering and assessments, and as informational documents intended for reviewers who are not directly involved in the program and who may not be familiar with the environmental disciplines requiring consideration.

The objectives of the planning function include:

- Ensuring a good correlation between data required and data available
- Ensuring a consistent program from region to region and contractor to contractor
- Ensuring complete coverage of total information needs that must be met in the environmental and geological study efforts
- Ensuring the availability of data necessary for site-qualification activities and licensing-document preparation.

Region specific survey plan documents were prepared for seven rock-salt regions for six states. Appendix volumes containing the external review comments and responses to those comments were also published. A generic (i.e., for all rock types and for all regions) regional survey plan is in preparation. This generic regional plan will include for each parameter addressed *rationales, data sources* for all regions of the country, and recommendations regarding *data presentation.* It is planned that generic area and location survey plans will also be prepared; these plans will be similar to the previous plans except that the guidance relative to *data sources* will be generic rather than specific.

GUIDES FOR PREPARING NEPA DOCUMENTS

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The Energy Reorganization Act of 1974 assigns responsiblity for siting, designing, and operating facilities for the terminal storage of nuclear wastes from commercial operations to the U.S. Department of Energy (DOE). This act also gives the U.S. Nuclear Regulatory Commission regulatory authority over such facilities. The national policy to "encourage productive and enjoyable harmony between man and his environment" is presented in the National Environmental Policy Act of 1969 (NEPA); applicable regulations for implementation of NEPA include the CEQ (Council on Environmental Quality) Regulations (40 CFR Parts 1500 - 1508, amended January 3, 1979) and the DOE Proposed Guidelines for Compliance (FR 44 pp. 42135 - 42147, July 18, 1979). Additionally, Executive Order 11752 (December 17, 1973), "Prevention, Control and Abatement of Environmental Pollution at Federal Facilities", requires compliance with federal, state, interstate, and local substantive standards and limitations.

The status of national policy with regard to nuclear waste management is presented in the final report (TID-29442, March, 1979) of the Intergency Review Group (IRG). The findings of this group as well as aspects of policy not defined by IRG will influence environmental assessments, particularly with regard to alternative considerations (i.e., Strategy I or II in which rock salt or other media for which the information is available would be considered for the first repositories, as opposed to Strategy III which would require the consideration of sites in three to five geologic media), the possibility of regional repositories, state and local "consultation and concurrence" (e.g., the possibility of a State Planning Council which would "provide state perspectives for the development of the National Nuclear Waste Management Plan, the site characterization programs . . ." and "assist DOE and the states in recommending proposed sites . . ."), and "public participation".

Proposed NRC procedures for exercising its regulatory responsibility with regard to health and safety and environmental protection matters would include safety review by the Atomic Safety and Licensing Board (ASLB) and environmental review for compliance with NEPA [Section 102(2)C]. It is anticipated that the NRC environmental review would, for each and any specific proposed repository, culminate in a Draft Environmental Impact Statement (DEIS) by NRC, agency review of the DEIS, a Final Environmental Statement (FES), and public hearings. It is anticipated that NRC will need and require an Environmental Report as the basis for the environmental review.

In accordance with the principle of tiering (described in the CEQ guidelines), the DOE has produced a generic DEIS on "Management of Commercially Generated Radioactive Waste" (DOE/EIS - 0046-D, April, 1979). This document considers and compares 10 alternative methods for disposal of nuclear wastes and concludes that the "state of technology stands out as a major decision factor, and the geologic disposal option has an edge over other options . . .".

The environmental assessment for particular repositories is now expected to begin with a Site-Recommendation Report (SRR). The primary input to the SRR would be the Site-Qualification Reports (SQR's) for a number of sites. The SRR would present a comparison of the sites and propose to advance one site for application for repository construction to the NRC.

The SRR will address the relative risks, impacts, and costs potentially associated with each of a small number of qualified candidate sites. The assessments will be based on generic (i.e., not-site-specific) engineering design information, detailed site data, including on-site meteorology and ecology data collected over a period of time (e.g., 6 months or more), and conceptual designs of mitigation measures. Impact-assessment models and calculation methods will only be as sophisticated as the accuracy of the input information justifies; assessments of the impacts of accidents and of low-probability long-term events will be included.

A plan has been prepared for the preparation of a DEIS which would (essentially) incorporate the SRR and add presentations of the purpose and need of the facility, a summary of the alternative technologies from the GEIS in order to systematically address each of the five considerations listed in Section 102(2)(C) of NEPA with regard to impacts, commitments, and alternatives. However, in view of the availability of the GEIS, the SRR, and the requirement of an ER to NRC, such a DEIS may not be required.

The ER must include detailed data and calculation methods to allow the NRC to perform a detailed, independent review. The site description will include a full year of data; the facility description will be site-specific preliminary design information; site-specific mitigation measures will be developed and presented; impact-assessments methods (i.e., models, computer codes, etc.) will either be presented or summarized with reference to other documents. Proposed programs for measurements and monitoring will be developed and discussed in relation to projected impacts to be monitored or other rationales. The repository need and consideration of alternatives will be addressed, but only briefly by summary and reference.

In each of the above-mentioned environmental-impact-assessment reports the short-term impacts refer to those incurred before the completion of decommissioning; the long-term impacts occur after decommissioning. Short-term impacts will include those due to construction, operation, transportation (of nuclear wastes and of excavation material), decommissioning, and abnormal events. Long-term impacts will be discussed in two time frames: the first period is the 100-1000 years after decommissioning and the second period is that which follows. The first period is characterized by considerably more radioactivity in the wastes (hence greater impact potential) and by the possibility of considerably more accurate environmental projections (i.e., climatic, ecosystem, societal structure, etc.) than the second period. The assessments of occurrences during the first period are more important and can be done with greater accuracy than for the second period.

It is anticipated that an acceptable license application will be required to demonstrate that safety and health impacts—even remotely possible long-term as well as short-term impacts—are minimal.

Guides for the preparation of SRR's, DEIS's, and ER's have been drafted. The guides will help ensure good management control and a uniform content of reports of the same kind for different sites. Moreover, the guides are a vehicle for coordination with other federal as well as state and local agencies; the guides also provide a vehicle for state and local consulation and concurrence and for public participation as required by the IRG guidelines. It is anticipated that during FY 1980 these plans will be revised to insure consistency with possible revisions in licensing planning prior to their earliest possible use in FY 1981.

ENVIRONMENTAL CRITERIA FOR IDENTIFYING SITES FOR GEOLOGIC REPOSITORIES

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The most visible activity of participants in the National Waste Terminal Storage (NWTS) Program is accumulation of data for use in site selection. The types of information being collected include geological, hydrological, meteorological, environmental, and socioeconomic data. To bring a single focus to such diverse information, and to establish an explicit site-selection objective, a system of site-selection criteria is required to guide siting investigations.

The objectives of site-selection criteria are several:

- Establish a consistent site-selection strategy
- Establish a consistent rationale for site-characterization data requirements
- Establish effective means for use of characterization data.

Meeting these objectives requires that the criteria possess certain characteristics, including:

- Consistent objectives throughout the criteria system
- Congruence with characterization data
- Specificity sufficient to make each criterion useful
- Flexibility sufficient to utilize the detailed characterization data.

Criteria Development

In December, 1978, an effort was initiated to consolidate and document site-selection criteria to be used by NWTS/ONWI. The approach taken was to screen the United States on the basis of reconnaissance-level information, a case distinct from comparing sites on basis of field data. The screening criteria would be statements of general objectives. These criteria would be supported by explicit identification of factors to be considered to show compliance with each criterion.

It was later decided that criteria at this level should be consistent throughout the NWTS Program. Factors and specifications necessary for comparing sites and for designing specific facilities would be the responsibility of each NWTS Program participant. Methods to be used in applying the criteria would also be left to the discretion of each NWTS component.

Screening criteria and their application to site selection are currently being agreed to by NWTS components. The comparative methodology, to be used for the first time about a year from now, is still under consideration. Alternatives being considered are rating-weighting, formal decision analysis, and modeling based on rudimentary impact assessment. A formal comparative methodology selection process for NWTS/ONWI has been proposed and is being considered for implementation.

Application of Criteria

Areas for further study have been identified within study regions using reconnaissance-level information and the screening criteria. The methodology selected as appropriate to this level of study has been the multiple-overlay approach. Conclusions and recommendations resulting from this procedure are documented in Summary Reports. It is in these reports also that the criteria and methodology are discussed in detail as they relate to characterization data for a particular region.

ENVIRONMENTAL SURVEYS OF THE GULF INTERIOR REGION AND THE PARADOX BASIN

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The goals of the Regulatory Project Manager (RPM) are to provide the necessary environmental data base, ensure compliance with the National Environmental Policy Act (NEPA), and generate the documentation that fulfills the regulatory requirements leading to site selection, licensing, construction, operation and closure of NWTS facilities for permanent underground disposal of spent fuel, and high-level and TRU wastes. These goals are provided through a study and reporting procedure consisting of:

(1) **Regional Studies**: gathering regional environmental data according to specific plans

(2) **Regional Summary and Area Selection**: identifying areas for further study by integrating regional environmental data with geologic data and criteria generated by other contractors

(3) **Area Studies:** gathering area environmental data according to specific plans

(4) **Area Summary and Location Selection**: identifying locations for further study by integrating area environmenta data with geologic data and criteria generated by other contractors

(5) **Location Studies**: performing full-year environmental, socioeconomic, and cultural surveys to acquire baseline data on locations

(6) **Location Summary and Candidate Site Selection:** recommending site(s) for licensing by preparing location data summary and site-qualification report, and applying criteria

(7) **Regulatory Documents**: preparing DER, PSAR, and DEIS for selected site(s).

In the Gulf Interior Region (Figure 1), BNI has been working closely with NWTS/ONWI and the NWTS geologic contractors (LETCo) to identify study areas in the Mississippi, Louisiana, and Texas interior salt-dome basins. In the regional study, data were gathered in the fields of meteorology, air quality, ecology, land use, agriculture, and regional economics, using existing literature and state and federal documents. These data, along with preliminary geologic and hydrologic data, were compiled in a Final Draft Regional Report issued November, 1978. Comments have been solicited from public groups and agencies in the region, and these comments are now being addressed.

The environmental data and geologic data provided by LETCo were summarized, and area selection criteria developed by ONWI were applied to these data. Criteria covered (1) depth, (2) size, (3) present use of domes (gas storage, mining, etc.), (4) presence of urban centers, and (5) presence of bodies of surface water. As a result of this synthesis, a Final Draft Summary Report was issued in May, 1979. Three areas (Figure 2) were selected for further study in that report, one in each state, each of which included at least two potentially acceptable domes. Approval to begin area studies was given in June, 1979, for Mississippi and Louisiana areas after public meetings were held in each state. Approval was given for start of Texas studies in July, 1979. Draft area reports are scheduled for issuance in early FY1980.

In the Paradox basin of Utah (Figure 3), BNI has worked with NWTS/ONWI and Woodward-Clyde to identify study areas. The Draft Regional Report was issued in May, 1978, for review, and the resultant comments are now being addressed. A Preliminary Draft Summary Report was issued in May, 1979, presenting geologic and environmental data and criteria for area selection. In the Paradox basin, selection criteria included: (l) depth to salt, (2) thickness of salt, (3) presence of faults, (4) presence of igneous features, (5) groundwater discharge, (6) presence of bodies of surface water, (7) energy and mineral resources, (8) presence of urban centers, and (9) present designated surface use (wilderness, parks, or monuments). Potential study-area boundaries are now being defined (Figure 3) , and authorization to gather area data was issued in July, 1979 . As in the Gulf Interior Region, a draft report on these areas is scheduled for issuance in early FY1980.

Also in FY1979, BNI began preparing for the year-long field programs required in location studies. Working with ERT, a subcontractor, BNI developed a field procedures manual, explored access to potential locations, and specified and costed field equipment, such as meteorological towers. The present schedule calls for field work to commence in FY1980 (March, 1980), after completion of area studies, issuance of Summary Reports, and responding to comments regarding location selection.

FIGURE 1. SALT DOMES AND ANTICLINE REGIONS IN THE UNITED STATES

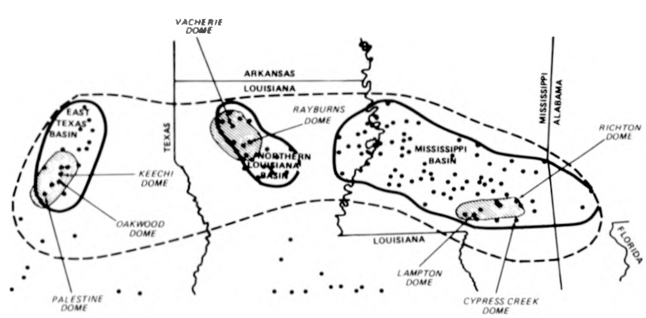

FIGURE 2. RECOMMENDED STUDY AREAS AND DOMES OF PRINCIPAL INTEREST IN THE GULF INTERIOR REGION

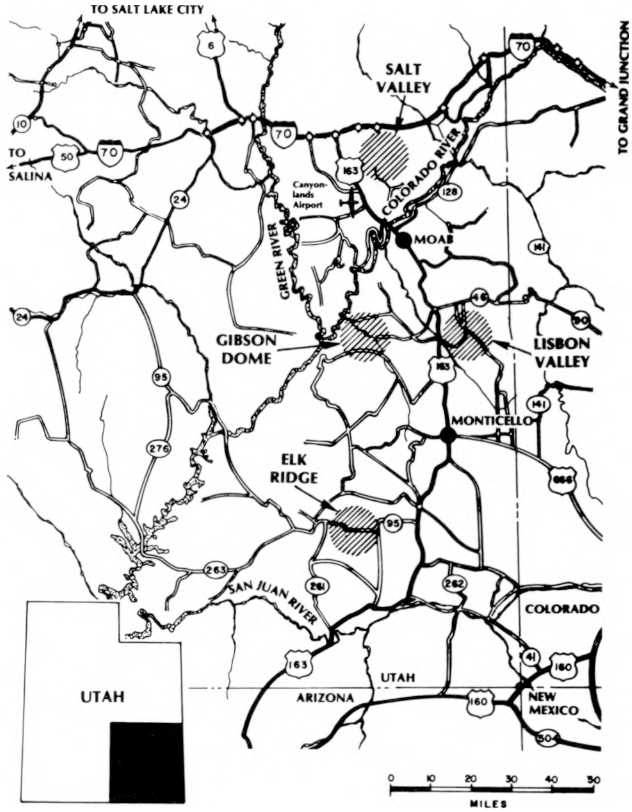

FIGURE 3. AREAS OF INTEREST IN THE PARADOX BASIN SALT DEPOSIT

ENVIRONMENTAL SURVEYS OF THE PERMIAN AND SALINA BASINS

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NUS Corporation, under contract with ONWI, is serving as one of the Regulatory Program Managers in support of the NWTS/ONWI site qualification and licensing program.The purpose of the work undertaken by NUS is to conduct environmentally related studies and prepare environmental/safety-related reports as required for review and licensing of an underground waste repository. The objectives of these studies are to ensure that the construction and operation of the waste repositories will be in full compliance with the requirements of the National Environmental Policy Act (NEPA) of 1969, the Code of Federal Regulations (CFR), and all other applicable federal, state, and local guidelines and regulations.

The basic studies to be performed include:

- TASK A: Regional characterization of the Permian rock-salt formation and the Salina salt formation*
- TASK B: Area studies within the Permian and Salina basins
- TASK C: Location studies within the Permian and Salina basins

*All studies relevant to the Salina Basin have been recessed pending resolution of Federal/State questions on site screening procedures.

- TASK D: Preparation of a complete Preliminary Draft Environmental Impact Statement (PDEIS) for a specific site
- TASK E: Preparation of a complete Environmental Report (ER) and Preliminary Safety Analysis Report (PSAR) for a specific site
- TASKS F AND G: Licensing support as required, including expert testimony related to the PDEIS, ER, and PSAR
- TASK H: Contribute to preparation of a Preliminary Information Report, using a reference repository site and and reference repository design.

The environmental studies under the RPM assignment are being focused by geologic investigations for potentially suitable geohydrological settings for repository locations. These geologic investigations are being performed separately from the RPM effort.

Regional Studies

Formation-wide, reconnaissance-level environmental characterization studies have been completed for both the Permian and Salina basins. The Permian and Salina basins are regions of bedded salt. The outlines of these regions are shown by Figures 1 and 2. Results of these efforts have been incorporated into two draft environmental characterization reports. One is under technical review. The other has been issued as ONWI-16 (Draft) "Environmental Characterization of Bedded Salt Formation and Overlying Areas of the Salina Basin", March, 1979.

In these characterization reports, the nature and use of the resources and overlying land areas of the bedded-salt regions are described in terms of environmental media (the geosphere, hydrosphere, and atmosphere) and natural background radiation. Also included is coverage that pertains to human (socioeconomic and land use) activities and natural (ecological) settings. The following represents a selection of interesting environmental observations extracted from the report summaries:

(1) Permian Basin

- The region encompasses approximately 189,000 square miles.
- The Permian region experienced a complex tectonic history during the Pennsylvanian period some 310 to 280 million years ago. Structural readjustments since that time have had little effect on the post-Permian rock units, including the salt sequences. The entire Permian region is seismically stable (seismic Zone 1).
- The Permian region has long been one of the major oil-and gas-producing regions of the United States.
- Groundwater represents a major resource in the region. Greater than 60 percent of the water withdrawn comes from groundwater. The largest single use of water in the region is for agriculture, which accounts for approximately 84 percent of the total consumption.
- Glaciers did not extend to the Permian region.
- Data indicate the national ambient air quality secondary standards for particulates are being exceeded throughout the western portions of the region and in some eastern areas.
- The Permian region is relatively sparsely populated. Only three urban areas in the region support a population of more than 100,000 inhabitants.
- Total earnings in 1970 were approximately 11 billion dollars. Agriculture, forestry, and fisheries accounted for approximately 14 percent; mining and extractive industries, 5 percent; retail, wholesale trade, government, and institutions accounted for 68 percent.
- The major land use is agricultural with 98 percent of the area classified as range or pasture (58 percent) and cropland (40 percent).
- A large portion of the region is semiarid with intermittent streams as the only aquatic habitat.

(2) Salina Basin

- The region encompasses approximately 80,000 square miles.
- Salt beds of the Salina basin were formed during the Paleozoic Era (400 million years ago).
- Major geologic structures of the region are extremely old; there have been no major movements in the earth's crust for approximately 180 million years.
- The region is one of low seismicity. Such earthquakes as have occurred in the Eastern region are attributed to readjustment of the earth's crust after the most recent ice age.

- Oil and gas fields have been developed in many parts of the region. Major bituminous coal fields occur throughout several of the states in the region.
- The region encompasses major hydrologic resources, both surface and groundwater. Surface water uses predominate due to a rainfall that ranges from 28 to 45 inches annually.
- Wind and precipitation patterns indicate low erosion potential in the region.
- Fundamental changes in climate have occurred over the last million years. During this period there have been four ice ages during which glaciers covered much of the region. The most recent ice age ended about 10,000 years ago.
- Many areas within the region are highly urbanized.
- Major land classifications of the region are forest land (44 percent), cropland (31 percent), pastureland (6 percent), and other rural land (6 percent).
- Total earnings in the region for 1970 are reported as 66 billion dollars. Manufacturing accounted for 41 percent of this amount. Retail and wholesale trade and government, institutional, and other services account for 56 percent.
- The Great Lakes and other regional surfacewater resources represent significant habitats for fish and other aquatic life forms.

Area Studies

Regional reconnaissance-level geological and environmental studies have been completed in both the Permian and Salina basins. Areas having features of particular interest were identified for further, more detailed studies. Area environmental-characterization studies in the Palo Duro and Dalhart subbasins portions of the Permian basin were initiated in August, 1979. The following provides a brief scope of the Environmental Study Program to be conducted in these subbasins.

The desired information will be obtained from the existing literature and by consultation with specialists/experts in the local areas. Information of interest includes:

(1) A description of the physical characteristics of the hydrosphere, including surface water and river basins

(2) A general characterization of the climate in the study area, including temperature range, precipitation, general air quality, and the occurrence of various weather events such as thunderstorms and hurricanes

(3) Demographic, socioeconomic, and land-use information to provide a characterization of the areas in terms of population densities, urban place locations, major transportation routes, and land-use patterns

(4) Terrestrial and aquatic ecosystems data, including rare and endangered species, unique habitats, major agricultural crops, and recreation areas.

Area environmental-characterization studies in the Salina basin have been delayed, pending resolution of federal/state questions on the site-search program.

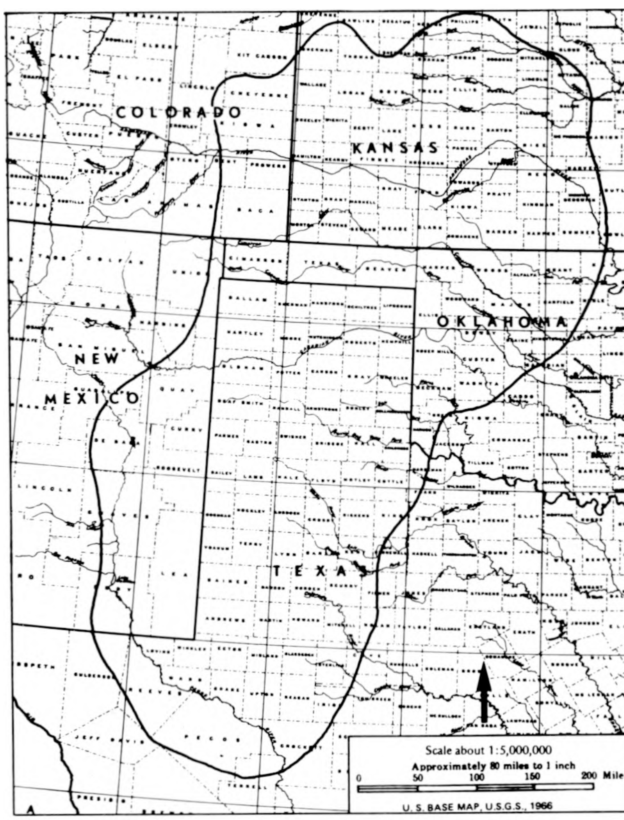
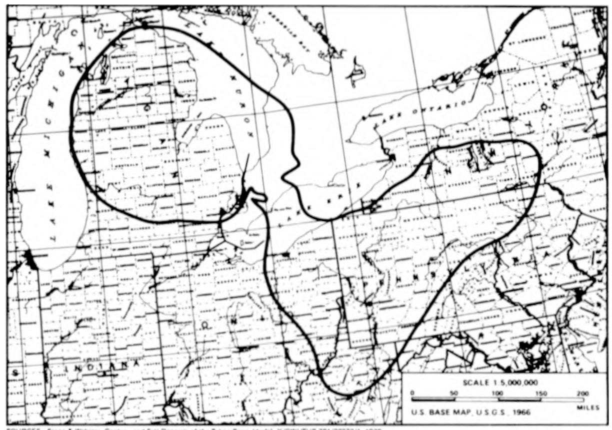


FIGURE 1. THE PERMIAN BEDDED-SALT REGION

FIGURE 2. THE SALINA BEDDED-SALT REGION



SOURCES: Stone & Webster, Geology and Salt Deposits of the Salina Basin (draft), Y/OWI/SUB 781/22332/1, 1978 and ERDA, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, ERDA 76-43, Vol. 5.

A PRELIMINARY INFORMATION REPORT FOR A GEOLOGIC NUCLEAR WASTE REPOSITORY

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The Department of Energy, in determining its strategy for site selection and licensing of a geologic repository for nuclear waste, is preparing a Preliminary Information Report (PIR) for submission to the Nuclear Regulatory Commission. The format and content will be similar to that expected to be necessary for a Preliminary Safety Analysis Report (PSAR) required to support an application to construct a repository. The content, however, will be directed more toward presentation of what information is presently known and what information needs to be developed for a complete PSAR. The PIR will describe the programs that will provide the input necessary for a full license application. The report will also identify how much information and analysis will be required to indicate safe isolation of radioactive wastes in a geologic medium and will delineate the effects of present knowledge and needs on overall NWTS program planning and goals. The purpose of the PIR is to provide a vehicle for integrating the technical programs from a licensing perspective, to determine where additional emphasis may be required to obtain high-confidence safety findings, and to aid in identifying and resolving licensing issues. An additional planned use of the PIR is to promote meaningful discussions between the Nuclear Regulatory Commission and the Department of Energy in regard to licensing a geologic repository.*

* Agreement between DOE and the NRC will be required with regard to the extent and output of any NRC review.

The PIR is based on a conceptual design for a repository in a reference geological/environmental regime. That is, the components of the system will represent a real design, a real surface environment, a real geologic-hydrologic regime, and a planned operation based on projected criteria and specification. The information presented will be specific to host medium, design, waste, environment, and analysis methodology, but not related to a specific site that might be selected for repository construction. The results given in the PIR will also be specific and will aid in identifying viable repository alternatives. The defined parameters will be used as envelopes for future evaluations in preparing a PSAR.

The PIR will be prepared using data, criteria, and analytical methods presently available from programs conducted by the Department of Energy. The report will present the state-of-the-art of repository design, operation, and decommissioning. Safety analyses are being performed to indicate the state of analytical methods and the status of models and criteria for near-term and far-term safety assessments for a proposed repository.

Report preparation is expected to take 15 months. Working papers were completed on June 30, 1979, and a working draft was produced on September 30, 1979. A preliminary draft to be completed on December 31, 1979, will be followed by a 3-month DOE review period. The conclusions and directions resulting from the review will be incorporated into a

June, 1980, second preliminary draft. The final DOE PIR will be published September 30, 1980. *

The report, if submitted to the Nuclear Regulatory Commission by DOE, may also be submitted to the ACRS and will be available to the public, interested states, and other government agencies. Document review and resulting discussions are expected to resolve many licensing issues and to identify specific issues that may require additional attention. The overall objective of the PIR is to focus the discussions of licensing a repository on those issues deemed most important. The results of the NRC review, if implemented, could be a preliminary safety evaluation report of the conceptual repository system. If reviewed by the ACRS, these could be a letter identifying unresolved concerns. The information gained from this review would lead to further definitions of major criteria and methodology and a better understanding of the important aspects of repository licensing.

*The June, 1980, and September, 1980, dates are contingent upon DOE direction to ONWI following the 3-month DOE review.

SESSION IIIC
TECHNICAL STUDIES IN THE NWTS/ONWI FACILITIES ENGINEERING PROGRAM

DESCRIPTION OF AN NWTS REPOSITORY FOR REPROCESSING WASTE IN DOMED SALT

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Stearns-Roger Engineering Corporation

In January, 1977, the Department of Energy initiated conceptual design activities, under the direction of OWI/ONWI, for federal repositories to permanently store radioactive wastes from the commercial nuclear power fuel cycle. Stearns-Roger developed a conceptual design, NWTSR1, for storage in domed salt of radioactive waste from the reprocessing of nuclear power-plant fuels. A second repository conceptual design, NWTSR2, was developed independently by Kaiser Engineers to store unreprocessed spent fuel in bedded salt.

The major purposes of the conceptual design activities were to develop defensible cost estimates, provide a head start on definitive design, and identify any research and development needed to provide the facilities and equipment required for a radioactive waste repository. Stearns-Roger has expended over 100,000 man-hours on the NWTSR1 design.

The repository is capable of storing all the reprocessing wastes generated through 2004 by a postulated high-growth U.S. nuclear power industry. Table 1 shows the principal parameters and Table 2 shows the schedule for NWTSR1. The majority of conceptual equipment items could be ordered today from vendor catalogs. Virtually all the remaining equipment requires engineering development only to adapt existing designs to the configurations and dimensions required for the repository.

The repository, shown in cutaway perspective in Figure 1, is divided into four major facilities. The Canistered Waste Facility handles high-level waste, cladding-hull waste, and intermediate-level-waste received in canisters.

The Low-Level TRU Waste Facility handles low-level TRU waste received in 55-gal drums. The Underground Facility encompasses two separated activities: (1) the canistered and low-level waste storage operations, and (2) the underground development operations. The Site Service Facilities encompass the various support facilities.

Site Service Facilities

The site, shown in Figure 2, covers approximately 2.1 km^2 (522 acres) on the surface. Site Service Facilities include buildings such as the administration and operations control center, mine operations building, and fan and filter building for the underground repository, installations to provide utilities such as electric power, and various other support facilities such as the backfill salt storage pile. Excess salt will be shipped off site for disposal.

Underground Facility

The cross section of the salt dome is assumed to be elliptical. The canistered waste storage area at the 600-m (2,000 ft) level is shown in Figure 3. The dome at this level is approximately 2.4 km (1.5 miles) wide and 4.8 km (3 miles) long. Low-level TRU waste is stored in a similar facility at the 450-m level. Six shafts are located in a central shaft pillar area: man and material shaft, low-level waste shaft, canistered waste shaft, two ventilation exhaust air shafts, and one ventilation inlet air shaft. Tower-mounted multiple-rope friction hoists are used in the major shafts.

High-level wastes generate sufficient decay heat to require spacing between canisters so as to avoid excessive temperatures. Therefore, the canistered waste storage area, when fully developed, will consist of 580 rooms spread over approximately 4.85 km^2 (1200 acres). A barrier pillar, approximately 245 m (800 ft) in width, extends between the mined area and the dome boundary. The low-level TRU waste storage area is similar to the canistered waste storage area, but has only 186 rooms due to the closer spacing permissible; low-level TRU waste generates essentially no decay heat. The ventilation system provides approximately 48,000 m^3/min (1.7 million cfm) of air to the underground facility. Flow through the development areas is always separated from that through the storage areas.

Canistered Waste Facility

Three types of waste are received at the Canistered Waste Facility. High-level waste is assumed to be a vitrified calcine in canisters 30.5 cm (12 in.) in diameter and 3.05 m (10 ft) high. Cladding-hull and intermediate-level wastes are assumed to be in canisters 61 cm (24 in.) in diameter and 3.05 m (10 ft) high. All waste canisters must be shielded and handled remotely because of their high radiation levels.

The Canistered Waste Facility, shown in Figure 4, consists of a shipping-cask inspection and decontamination area, a shipping-cask off-loading and handling area, hot cells for unloading and handling canisters, a transfer-cask loading area, and an area for loading transfer casks onto the hoist for lowering underground. Shipping casks are inspected, decontaminated if necessary, off-loaded from the railcars, and moved inside the Category I boundary, where they are interfaced with a hot cell. The canisters are removed from the shipping cask into the hot cells, where they are inspected for damage and radiation levels and overpacked if required. Canisters are loaded into transfer casks and lowered to the underground facility via the canistered waste shaft. Underground, a rubber-tired transporter moves the loaded transfer cask through the access drifts to the entrance of a storage room. A gantry crane moves the transfer cask to a predrilled disposal hole. The canister is then lowered into the hole and released. Empty transfer casks are returned to the surface for reloading. Empty shipping casks are inspected and returned to the shippers.

Low-Level TRU Waste Facility

Low-Level TRU waste is defined as waste suspected of containing at least 10 nanocuries of transuranics per g and having surface radiation levels of no more than 10 nrem/hr. These waste containers are handled unshielded, principally by fork-lift trucks. Low-level waste is assumed to be incinerated and solidified in concrete in 55-gal drums shipped in a cargo carrier by rail or truck.

Upon receipt at the Low-Level TRU Waste Facility, the cargo carriers are opened, and the drums are removed, inspected for damage and surface radiation levels, and overpacked if required. The drums are palletized and lowered underground via the low-level waste shaft hoist cage. They are then loaded on a rubber-tired transporter and moved to a storage room. Fork-lift trucks position the palletized drums in their final storage location. The empty cargo carriers are inspected and are returned to the shippers.

TABLE 1. NWTSR1 PRINCIPAL PARAMETERS

Parameter	Values
Nominal area	4.85 km^2 (1200 acres)
Nominal decay heat density	37.1 w/m^2 (150 kw/acre)
MTU equivalent stored	138,600 MTU
Duration waste receipts	20 years
Canistered Waste: Canistered	98,578
Mine level	-600 m (-2000 ft)
LLTRU waste: Drums	532,666
Mine level	-450 m (-1500 ft)
Salt excavated	19.2 million tons
Surface facility area	2.11 km^2 (522 acres)
Overall costs	<0.1 mill/kw-hr

TABLE 2. NWTSR1 SCHEDULE

Year	Activity
0	Select site, award engineering
2	Begin construction
8	Begin waste emplacement
13	End retrievability phase
27	End waste emplacement
31	Complete decommissioning

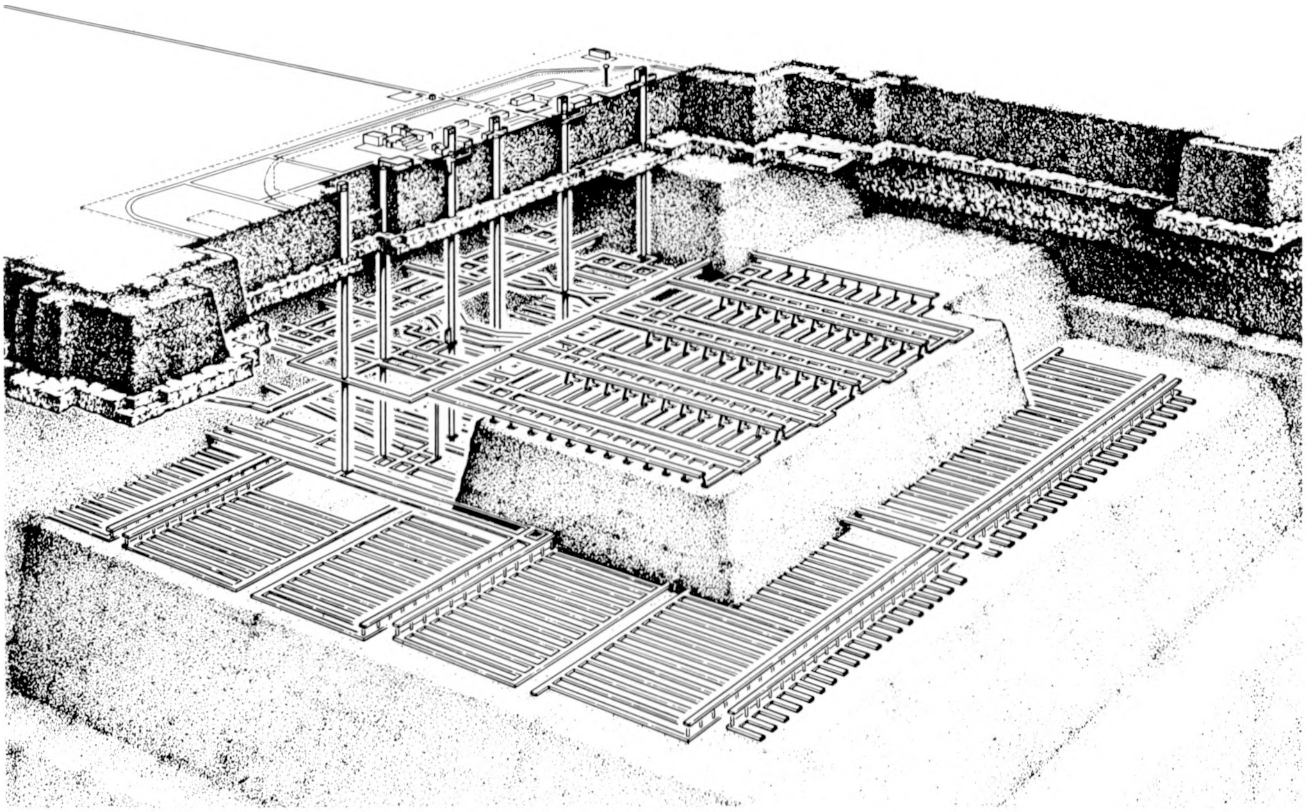
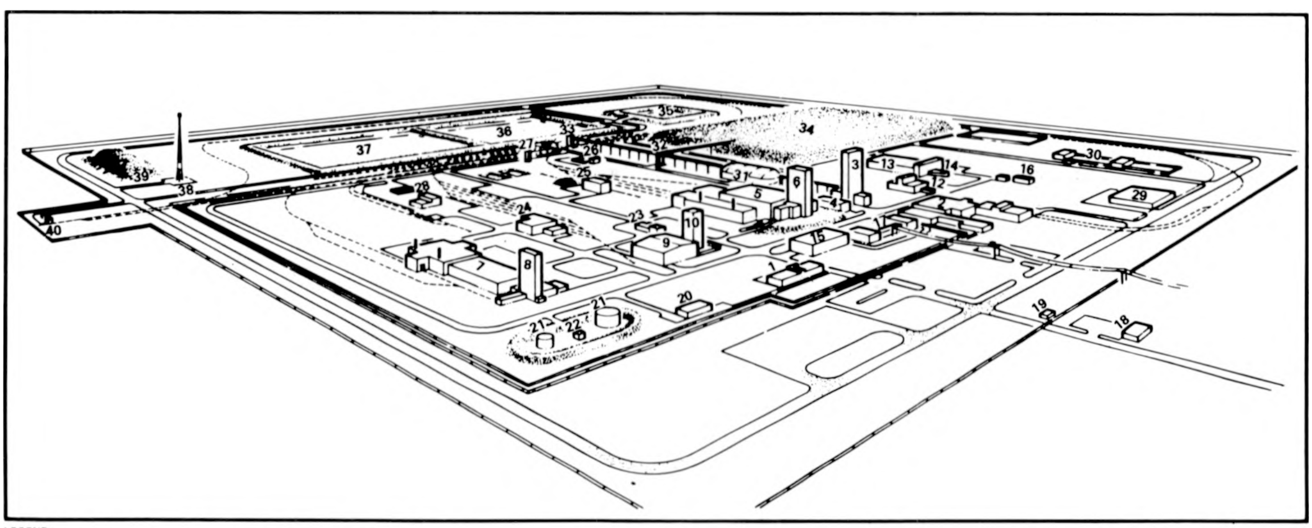


FIGURE 1. REPOSITORY PERSPECTIVE



LEGEND

1 ADMINISTRATION & OPERATIONS CONTROL CENTER
2 UNDERGROUND OPERATIONS BUILDING
3 MAN AND MATERIAL SHAFT HOIST TOWER
4 MAN AND MATERIAL SHAFT FILTER BUILDING
5 CANISTERED WASTE RECEIVING BUILDING
6 CANISTERED WASTE SHAFT HOIST TOWER
7 LOW LEVEL TRU WASTE FACILITY BUILDING
8 LOW LEVEL WASTE SHAFT HOIST TOWER
9 VENTILATION SUPPLY FILTER BUILDING
10 VENTILATION SUPPLY SHAFT HOIST TOWER
11 DEVELOPMENT EXHAUST FILTER BUILDING
12 DEVELOPMENT EXHAUST SHAFT
13 REPOSITORY EXHAUST FILTER BUILDING
14 REPOSITORY EXHAUST SHAFT
15 COMPRESSOR AND CHILLER BUILDING
16 COOLING TOWER
17 ELECTRICAL POWER BUILDING
18 VISITORS CENTER
19 ROADWAY GATEHOUSE
20 ENVIRONMENTAL AND INSTRUMENT LABS BUILDING
21 POTABLE WATER TANKS
22 FIRE PUMPHOUSE
23 FIREHOUSE, SECURITY AND MEDICAL CENTER
24 RADIOACTIVE WASTE FACILITY
25 STEAM GENERATING PLANT
26 SEWAGE TREATMENT PLANT
27 RR CONTROL TOWER
28 MAINTENANCE BUILDING
29 SITE WAREHOUSE
30 STORAGE SHEDS AND YARD
31 LIVE STORAGE AND SURGE BIN BUILDING
32 SALT TRANSFER TOWER
33 RAILROAD LOADOUT STRUCTURE
34 MINED SALT SURFACE STOCKPILE
35 STOCKPILE RUNOFF HOLDING POND
36 PROCESS WATER HOLDING POND
37 RETENTION POND
38 ENVIRONMENTAL MONITORING TOWER
39 ASH AND SLUDGE DISPOSAL AREA
40 RAILROAD GATEHOUSE

FIGURE 2. SITE PERSPECTIVE

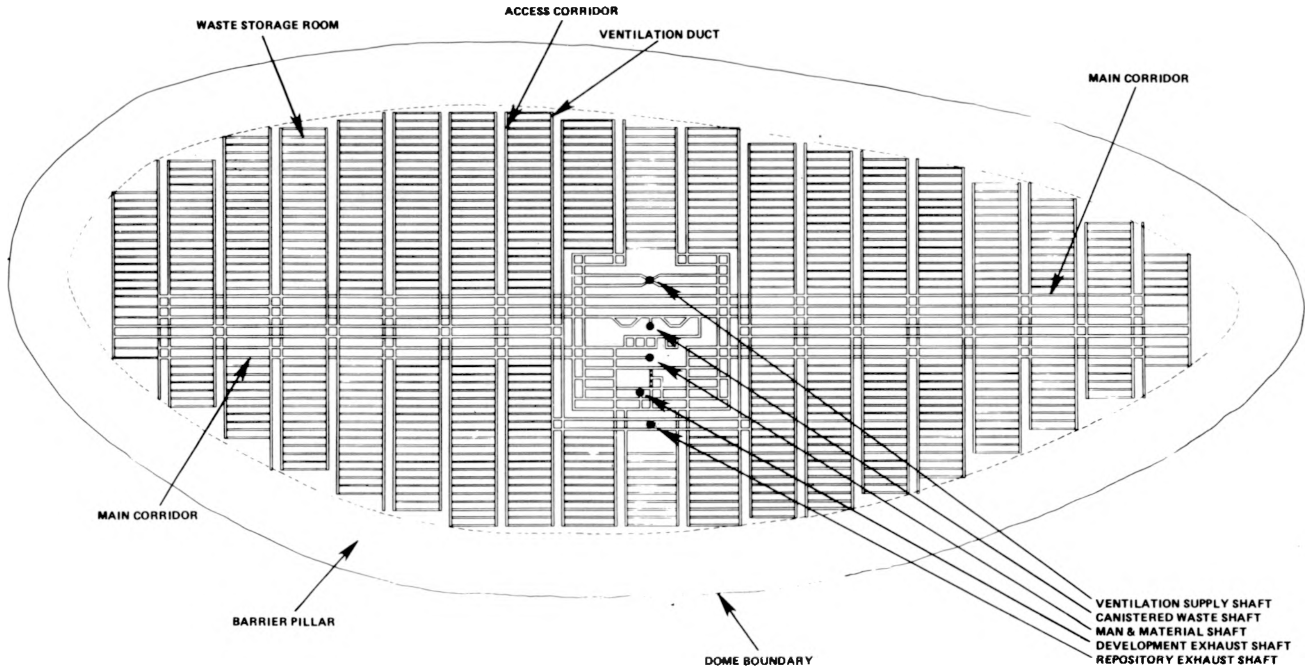


FIGURE 3. MINE PLAN

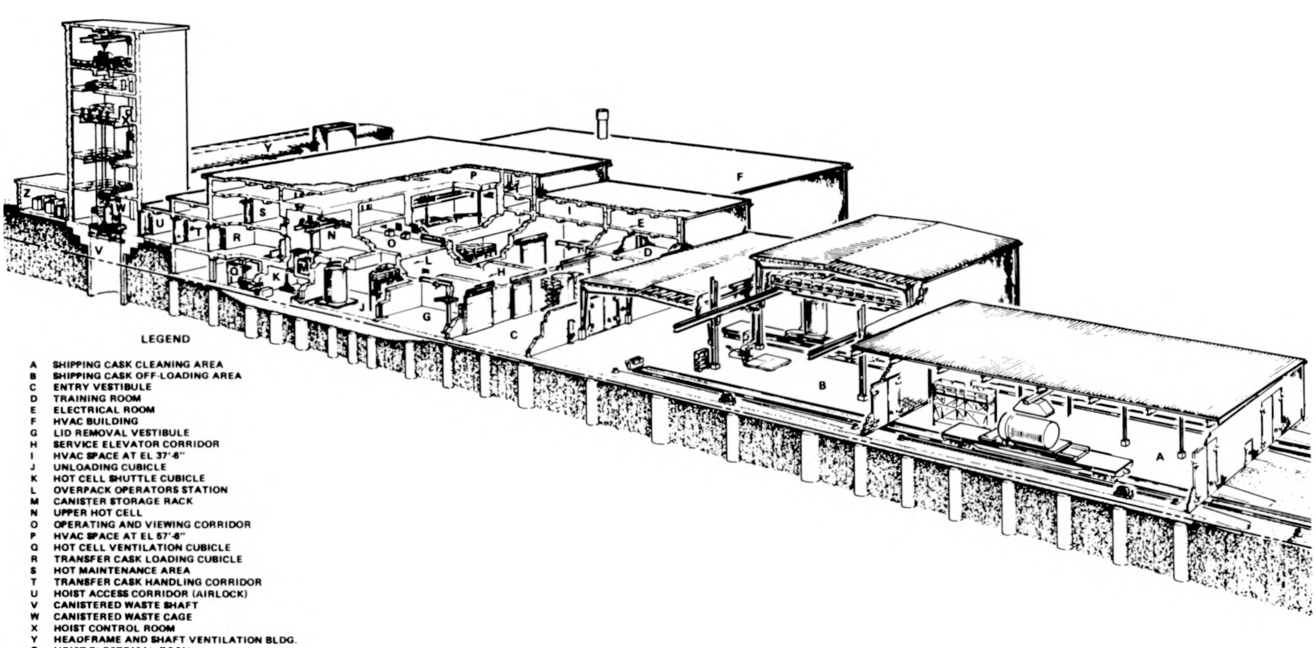


FIGURE 4. CANISTERED WASTE FACILITY SECTIONAL FACILITY

DESCRIPTION OF AN NWTS REPOSITORY FOR SPENT FUEL IN BEDDED SALT

J. S. Ritchie, E. A. Dukleth, J. S. Busch, and D. L. Watson
Kaiser Engineers, Inc.

The Conceptual Design of National Waste Terminal Storage Repository 2 (NWTSR2) was performed by Kaiser Engineers from January, 1977, through the 1979 calendar year under contract with the Department of Energy's Oak Ridge Operations Office. The NWTSR2 Conceptual Design is based on the storing of precanistered spent-fuel assemblies and low-level waste in a 2000-acre bedded-salt repository at a hypothetical site in the midwestern United States, at a depth of 2000 feet. The waste is to be stored so that it is retrievable for 5 years — retrievability is defined as requiring open rooms in the mine.

The spent fuel is to be 10 years old and to be stored at 60 kw per acre in a licensed facility. Canister receipts range from about 4000 per year during the 5 year retrievability period up to a maximum of 21,000 spent-fuel canisters per year after 22 years. Canisters contain one PWR or two BWR fuel assemblies. Table 1 summarizes the principal NWTSR2 parameters.

Figure 1 is an artist's conception of the repository showing a portion of the mine and showing four shafts connecting the mine with the surface facilities above. Figure 2 shows a perspective cutaway of the mine looking from the shaft pillar end. Figure 3 shows the design basis geologic column which was used for the conceptual design of the mine.

Although a repository in salt is often referred to as a "salt mine", Table 2 points out how the design objectives of NWTSR2 differ from those of a commercial salt mine.

Addressing the underground facilities briefly, Figure 4 shows an enlarged perspective of a portion of the shaft pillar area; Figures 5 and 6 show a typical storage room in the mine. Figure 7 shows cross-sections of the 4 shafts, which range in inside diameter from 10-1/2 ft. for the smallest to 22 ft. for the largest.

Figure 8 shows a typical continuous mining machine in operation in a Saskatchewan potash mine. This is the type of machine which is planned for NWTSR2.

One of the principal restraints in the design of a geologic repository is the maximum temperatures achieved in the geologic medium (salt in this case). Figure 9 shows the temperature profile between storage rooms at the end of the 5-year retrievability period.

Another key technical aspect of the geologic repository is the waste-handling function—from receipt of canistered fuel in shipping casks to emplacement in the mine and backfilling. Figure 10 lists the principal waste-handling functions. Figure 11 shows a portion of the surface facilities with the waste-handling building in the center.

The unloading of the shipping cask is shown in Figure 12. The cask is received horizontally and lifted to an upright position — then mated with the hot-cell unloading port.

Figure 13 shows the canister mover which lifts each canister from the shipping cask, moves it into the hot-cell, and then to the various positions in the hot-cell as shown in Figure 14. After a canister has been placed in the transfer cask, it is moved to the transfer handling equipment at the waste cage at the top of the waste-handling shaft. After the transfer cask has been lowered down the shaft, the canister is removed from the cage by the equipment, shown in Figure 15, and placed in the transporter horizontally as shown in that drawing. Figure 16 shows a larger view of the transporter, with the transfer cask in vertical position and showing it in phantom in horizontal position for transport.

Figure 17 shows a storage hole and the transporter during loading operation. All canister handling is by electronmagnets.

Some of the things learned from the conceptual design studies are listed below:

- Heat generation from implanted canisters does not limit access for storage or for retrieval operations.
- It is feasible to receive and handle canisters dry for both 10-year-old and occasional 3-year-old spent fuel.

- Transfer casks should be used while moving spent-fuel canisters from hot-cell to mine.
- Rapid canister unloading, handling, and transfer is feasible and economical using automated industrial type machines.
- Mine ventilation air does not require cooling for the assumed midcontinental site and negligible water inflow.
- Cool-down is not required for casks containing 10-year-old fuel assemblies.
- Proven, conventional mining methods are applicable and feasible for NWTSR2.

TABLE 1. MAJOR CRITERIA AND CONSTRAINTS

1. Store precanistered spent fuel and low-level waste
2. In bedded salt
3. Hypothetical site
4. Depth of 2000 feet
5. 2000-acre mine
6. Retrievable for 5 years
7. Open mine rooms
8. 60 kw/acre with 10-year-old fuel
9. Licensed facility

TABLE 2. REPOSITORY DESIGN OBJECTIVES DIFFERENCES FROM A COMMERCIAL MINE

Features	Mine	Repository
Depth	Near surface as economical	Depth sufficient for waste isolation
Surface penetrations	Restricted only by economics	Restricted to maintain repository integrity
Extraction ratio	Maximum to achieve low recovered ore cost	Minimum to provide mine integrity and low waste storage cost

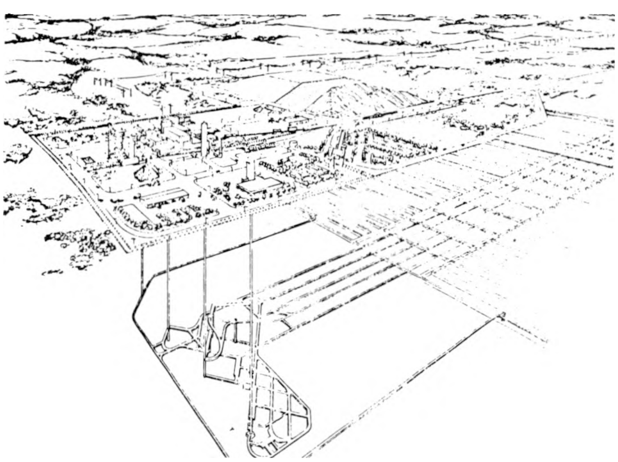


FIGURE 1. PERSPECTIVE VIEW OF NWTS BEDDED-SALT REPOSITORY

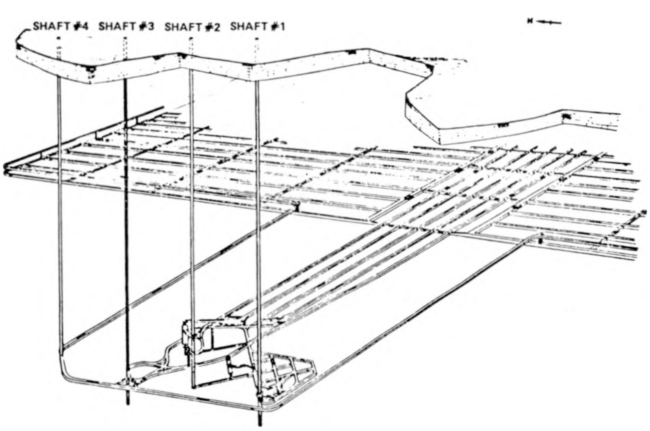


FIGURE 2. MINE LAYOUT PERSPECTIVE

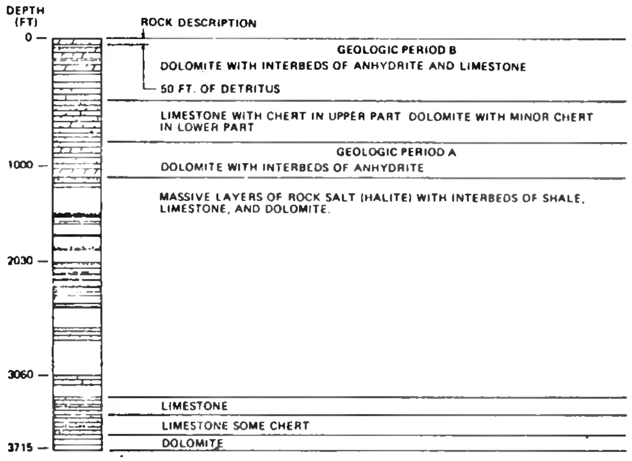


FIGURE 3. DESIGN BASIS GEOLOGIC COLUMN

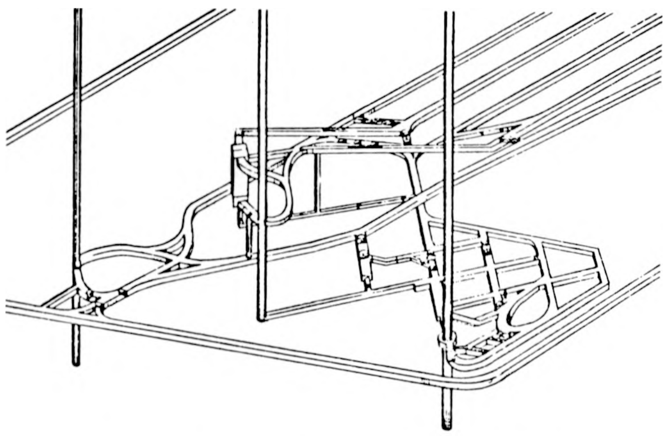


FIGURE 4. PILLAR LAYOUT PERSPECTIVE

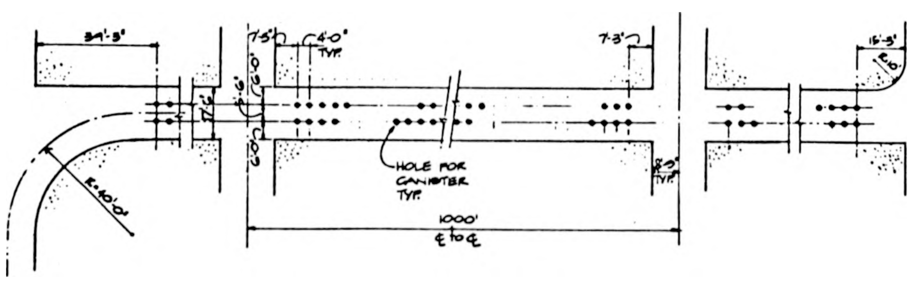


FIGURE 5. TYPICAL FUEL ASSEMBLY STORAGE ROOM

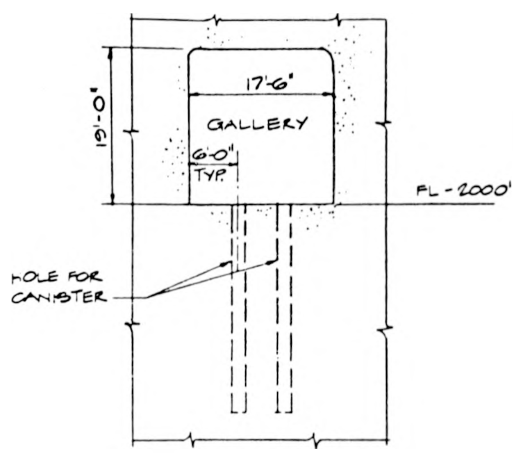


FIGURE 6. TYPICAL STORAGE ROOM

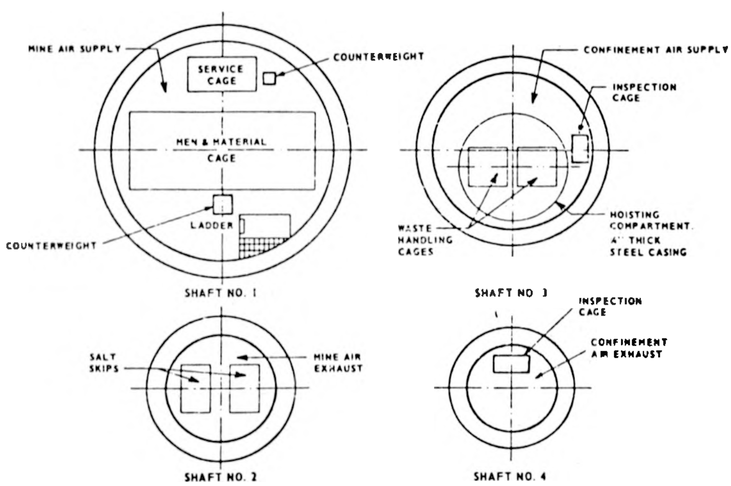


FIGURE 7. SHAFTS

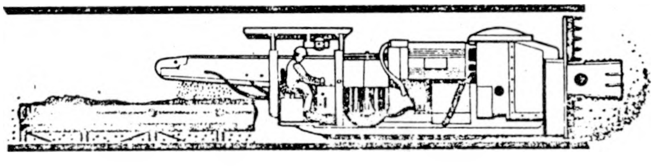


FIGURE 8. BORING TYPE CONTINUOUS MINING MACHINE WITH EXTENSIBLE BELT CONVEYOR

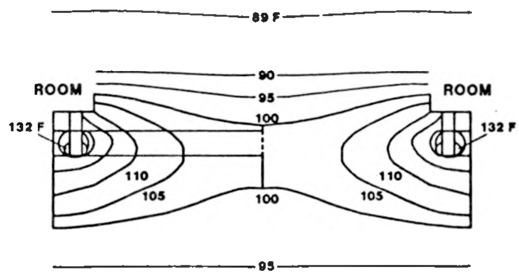


FIGURE 9. PILLAR TEMPERATURE PROFILES RETRIEVABLE MODE 5 YEARS AFTER STORAGE

1. RECEIVE AND INSPECT CANISTERED SPENT FUEL ASSEMBLIES
2. UNLOAD SHIPMENT -
3. TRANSFER CANISTERS TO MINE LEVEL
4. TRANSPORT CANISTERS TO REPOSITORY ROOM
5. PLACE CANISTERS IN STORAGE HOLE
6. RETRIEVABILITY FOR 5 YEARS

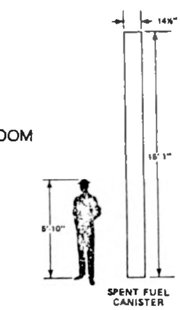


FIGURE 10. WASTE HANDLING FUNCTION

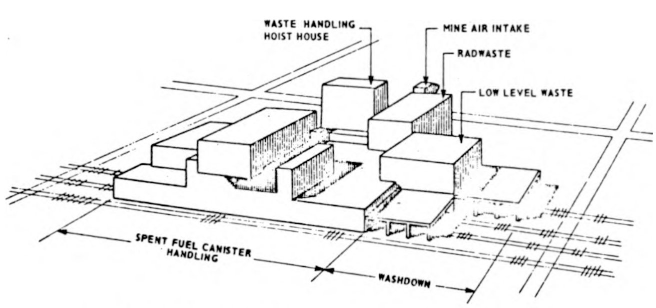


FIGURE 11. WASTE HANDLING BUILDING

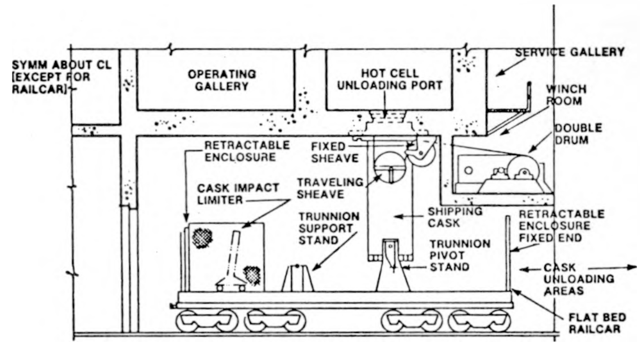


FIGURE 12. SHIPPING CASK UNLOADING EQUIPMENT ARRANGEMENT

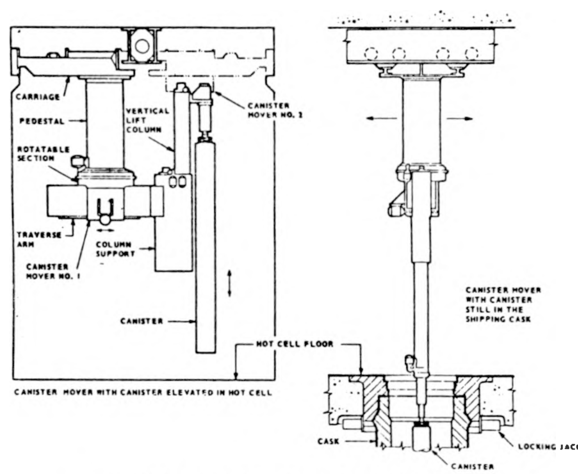


FIGURE 13. HOT CELL CANISTER MOVER

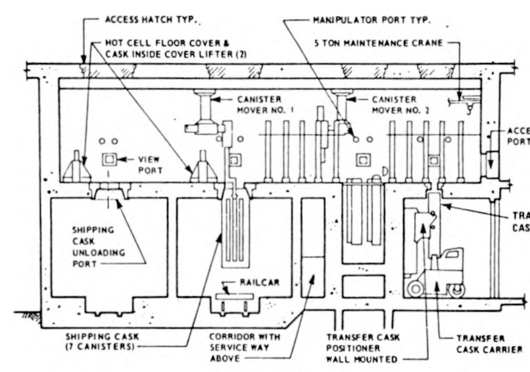


FIGURE 14. HOT CELL OPERATIONS

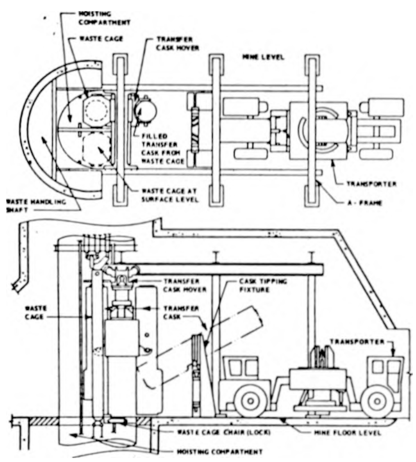


FIGURE 15. TRANSFER CASK HANDLING EQUIPMENT FROM WASTE CAGE

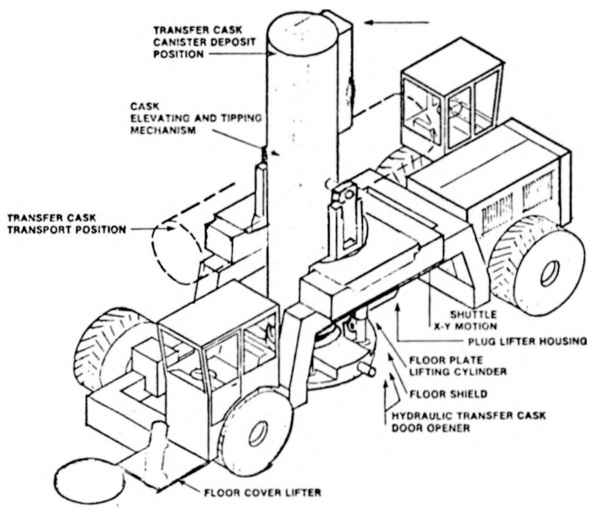


FIGURE 16. TRANSPORTER

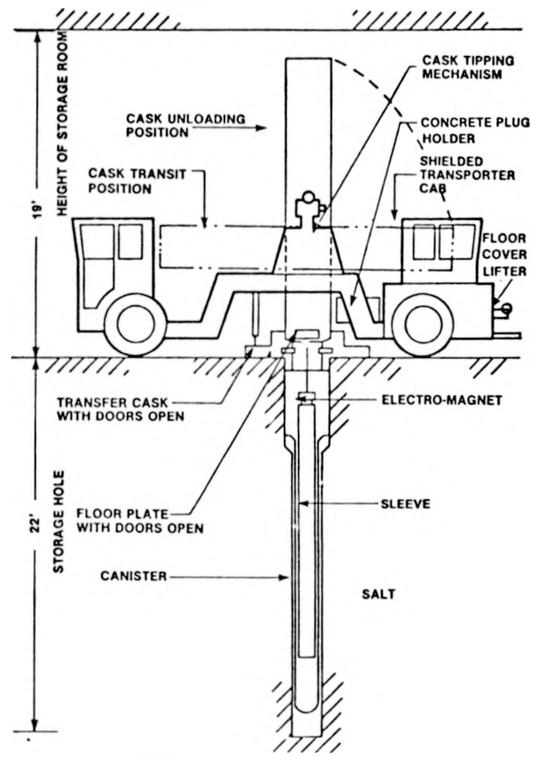


FIGURE 17. NWTS-2 TRANSPORTER & STORAGE CONCEPT

Dup

SPENT-FUEL RECEIVING AND PACKAGING FACILITY CONCEPTUAL DESIGN

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Rockwell Hanford Operations

F. L. Harris
Kaiser Engineers, Inc.

In October, 1976, and in April, 1977, presidential policy statements on nuclear energy were announced which included the indefinite postponement of reprocessing of light water reactor (LWR) spent fuel. The Spent Unreprocessed Fuel (SURF) Program was initiated to provide facilities to receive and package the spent fuel and to provide interim storage in surface structures with a passive cooling mode. This paper describes the conceptual design of the receiving and packaging facility module for that program. The design is also applicable to packaging spent fuel for geologic disposal. The design was produced by Kaiser Engineers under Rockwell Hanford Operations program management and published in RHO-CD-506 DRAFT, "Spent Fuel Receiving and Packaging Facility Conceptual Design Report", dated September, 1978.

The size of the facility is based on encapsulation within a 15-year period of all LWR spent fuel assemblies which will have accumulated as a result of commercial nuclear power plant operation through the year 1990. This resulted in a throughput of 3300 MTU/ year. The packages were designed to contain 1-PWR or 3-BWR fuel assemblies, which resulted in 1.0-kw heat output per package for spent fuel out of the reactor 5 years after a 33,000 Mwd/mt burnup for PWR fuel and 27,500 Mwd/mt burnup for BWR fuel. The facility will produce 27 packages per day, working three shifts in the receiving area 7 days a week for 300 days per year, and three shifts in the encapsulation area 5 days a week for 250 days per year.

The facility can receive fuel by rail or truck transport. The spent fuel is unloaded in a water pool and moved to a lag storage pool of 850 MTU capacity (25 percent of annual throughput).

The spent fuel is conveyed to the weld and test cell, where it is inserted into a canister and an end cap is mounted on the canister and held in place. A welding positioner rotates the canister to the horizontal position and a closure weld is made using remote handling equipment. A transfer jib hoist moves the canister between subsequent stations, where post-weld heat treatment can be conducted if needed, a nondestructive examination of the weld is performed (ultrasonic method), the canister is pressurized with helium, a helium leak check is conducted, and the outer surface is decontaminated and dried. The package can then be loaded out of the facility or put in a lag storage pit (air cooled) with capacity for 12 packages in each weld and test cell. One hundred percent redundance is provided with two weld and test cells, each capable of handling 100 percent of the throughput when in full operation.

The loadout mode is not defined in this design, but the spent-fuel packages could be loaded into transfer casks for direct transport into a geologic repository, transporters for movement to a surface storage facility, or rail or truck transportation casks.

Other features of the facility include a cask holding area for temporary storage of shipping casks and carriers to accommodate fluctuations in deliveries, cask maintenance stations for minor repairs, a cask cooling and washdown station, a cask decontamination pool, and provisions for isolating and storing canned leaking fuel assemblies. Also, a special-function cell is provided to receive and package damaged or leaking spent fuel, and to de-encapsulate spent fuel from defective containers for return to the production line.

The support building complex includes an administration building, security building, warehouse, maintenance building, emergency generator building, transport-vehicle maintenance shop and inspection pit, and boiler house. Utilities, roads, parking, and security fencing are also supplied.

The total capital cost of the facility is estimated at $400 million, and the annual operating cost—including wages for 410 personnel, maintenance, supplies, utilities, and 6665 canisters — is $48.5 million. The costs are expressed in 1978 dollars.

CONCEPTUAL REFERENCE REPOSITORY DESCRIPTION

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Bechtel National, Inc.

As part of the NWTS/ONWI program, Bechtel National, Inc., prepared a description of a conceptual repository for salt-dome terminal storage of unreprocessed spent fuel assemblies and low-level transuranic waste from commercial light water nuclear power reactors. This Conceptual Reference Repository Description (CRRD) is a compilation of features from the following: the Stearns-Roger conceptual design (Report NWTS-R1) for terminal storage of HLW in domed salt and the Kaiser Engineers conceptual design (Report NWTS-R2) for terminal storage of spent-fuel assemblies in bedded salt, and the Rockwell Hanford Operations conceptual design for a facility for receiving and packaging spent fuel assemblies (RHO-CD-50 Draft). This CRRD is not a conceptual design development as such, but combines features from these other reports to form a description of a repository for use as a basis for preparation of a Preliminary Information Report (PIR) for salt repositories. This Preliminary Information Report is to be submitted to DOE and, eventually, to the NRC. The CRRD and PIR will be followed by an actual site selection, and preparation of a repository conceptual report, a preliminary design report, and a preliminary safety analysis report.

The CRRD describes from a conceptual standpoint, the structures, systems, equipment, and operations necessary to receive unreprocessed spent fuel, place it in a carbon steel canister, and transfer and emplace the canister in the salt dome for terminal storage. The principal features of the conceptual design are shown in Figures 1 through 5.

The facility is based on providing regional storage, receiving approximately 260,000 BWR and PWR assemblies and 26,800 drums of low-level transuranic waste representing one-third of the national output from commercial reactors over the 30-year emplacement period.

The principal surface structure is the waste-handling building. Here, spent fuel assemblies are received in standard shipping casks via train or truck. The casks are off-loaded, and the fuel assemblies unloaded and temporarily stored under water. From wet lag storage, the assemblies are transferred into a hot cell where they are dried, placed in a carbon steel canister, and then transferred underground for terminal storage.

The underground repository contains approximately 2100 acres, which includes an 800-ft buffer zone between the storage area and the edge of the dome. There are five vertical shafts for transfer of men and materials, salt handling, waste handling, and ventilation supply and exhaust.

A shaft pillar area at the center of the repository beneath the surface facilities provides support for the shafts and encompasses the shaft stations and underground operations facilities. Main access and branch corridors connect the central pillar area to the 21-ft-high by 18-ft-wide by 540-ft-long storage rooms.

The canisters, contained inside a transfer cask for radiation shielding, are transported from the waste shaft to storage room by rubber-tired, diesel-driven transporters, picked up by a railmounted gantry crane, moved into the storage rooms, and emplaced in vertically drilled holes, 23 ft deep, spaced 4 ft center-to-center in two rows 5.5 ft apart.

Underground ventilation is provided by two independent systems, one for the emplacement area and one for the development area. The ventilation systems permit control of flow of air underground and prohibit any contamination releases due to underground accidents.

The facility is designed to accommodate retrieval of canisters starting at any time up to 5 years following initial emplacement, using the same basic equipment and operations as in the emplacement. Subsequent to retrieval, but prior to final shaft sealing, the canisters can be recovered using special equipment and techniques.

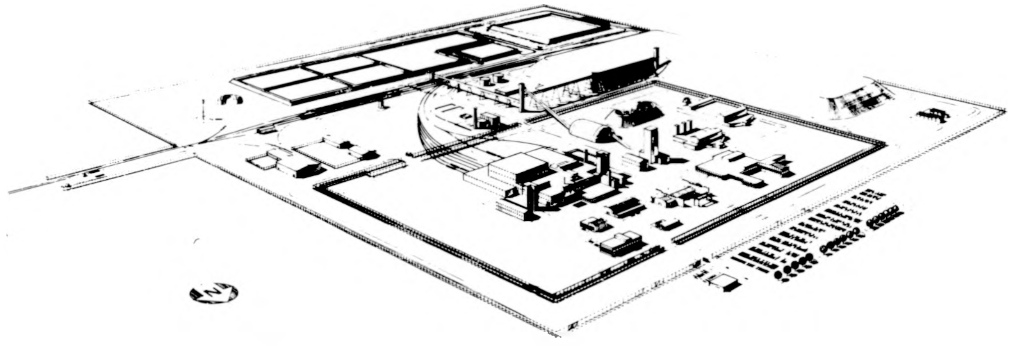

FIGURE 1. NWTS-CRRD PLANT FACILITIES PERSPECTIVE

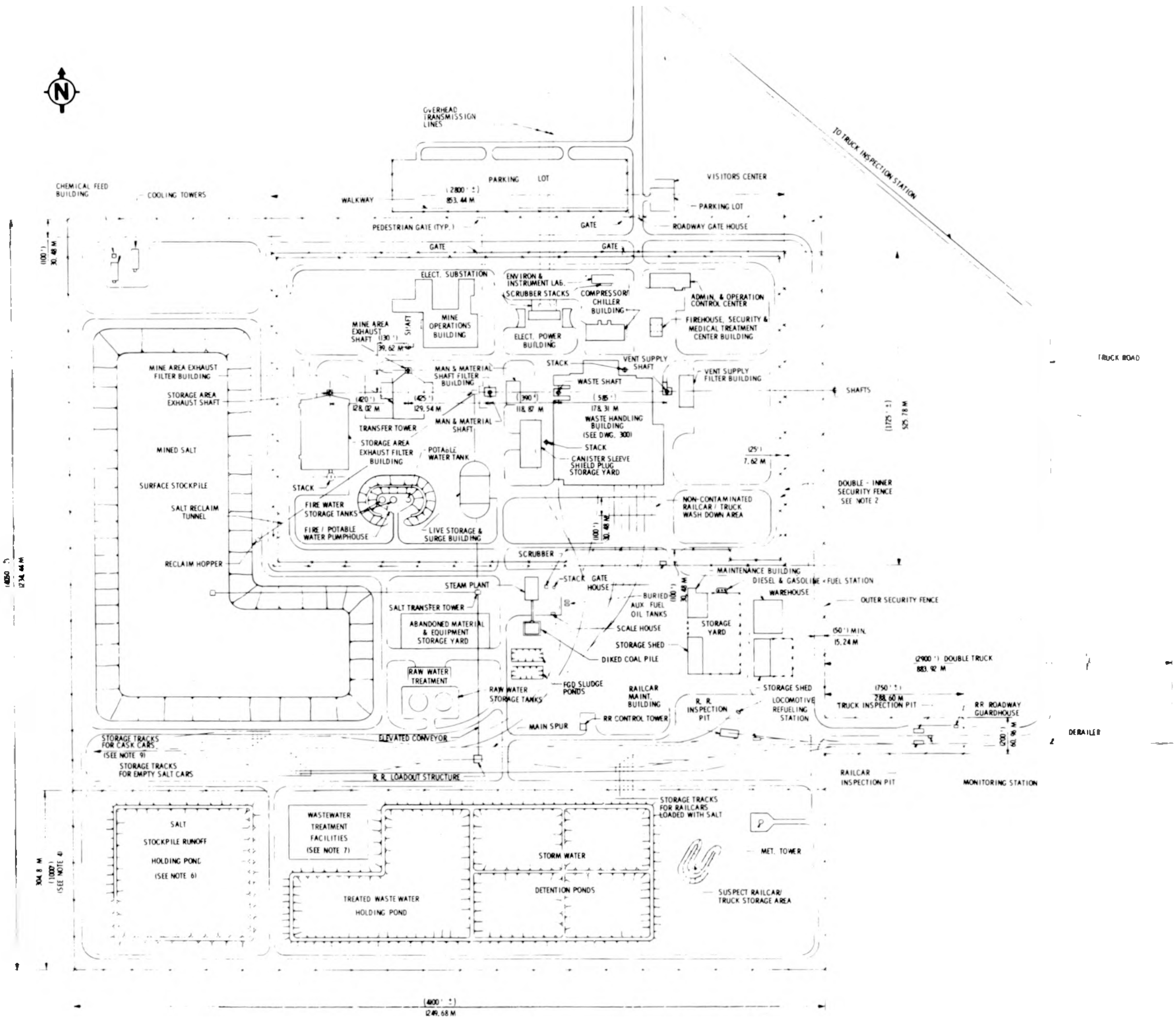

FIGURE 2. CRRD-PLOT PLAN

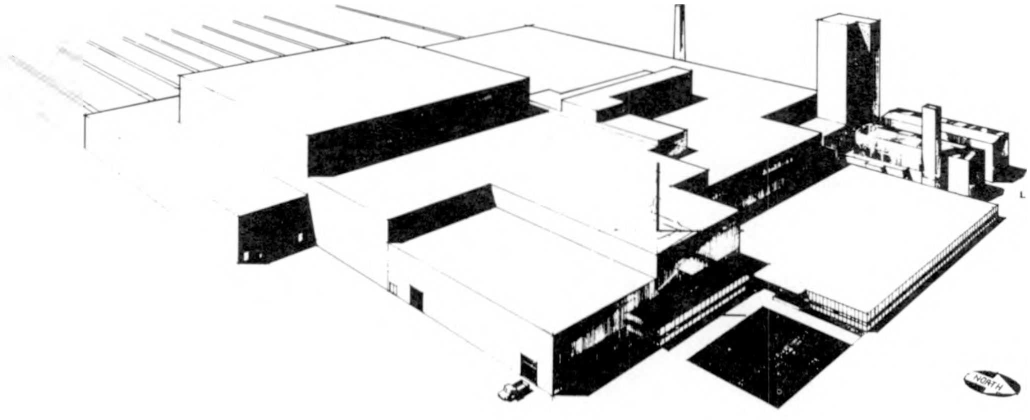

FIGURE 3. WASTE HANDLING BUILDING PERSPECTIVE

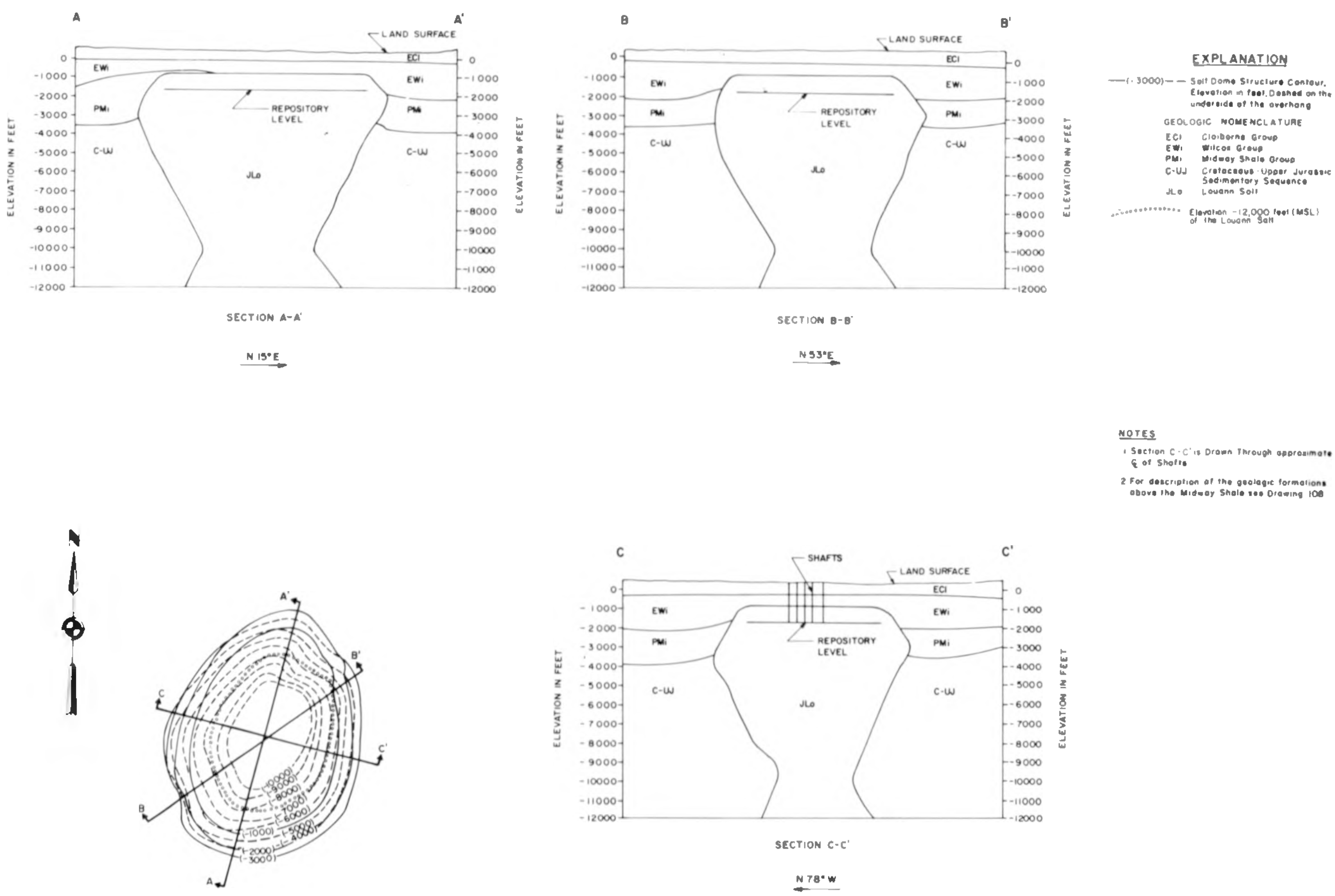

FIGURE 4. SALT DOME CONFIGURATION WITH GEOLOGY

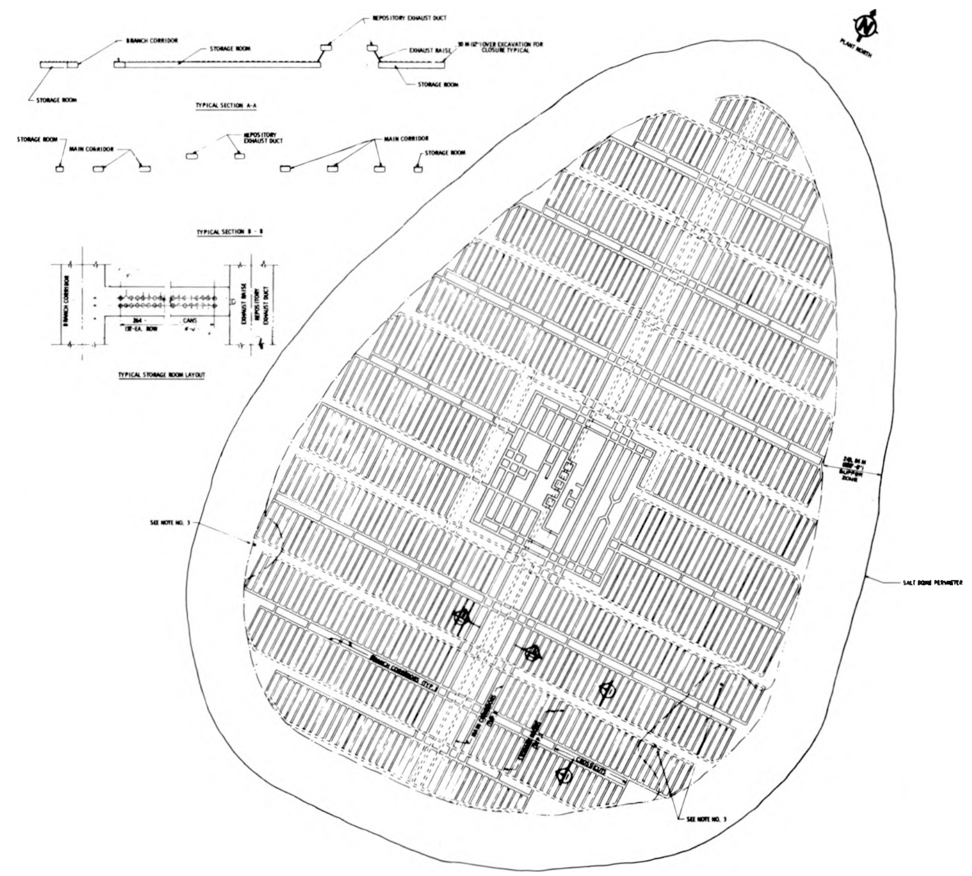

FIGURE 5. NWTS-CRRD UNDERGROUND LAYOUT

AN OVERVIEW OF INTERNATIONAL NUCLEAR FUEL-CYCLE EVALUATIONS (INFCE) SUPPORT

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A key element of the President's nuclear energy policy is a basic concern that the civilian nuclear power course chosen be as resistant as possible to proliferation of nuclear weapons—the obtaining of nuclear weapons capability by nonweapons states. Obviously this issue is international in scope and, furthermore, the realm of acceptable solutions lies in a combination of political, social, technical, and ethical considerations. To this end, systematic studies have been undertaken in both the national and international arena: the Nonproliferation Alternative Systems Assessment Program (NASAP) in the U.S. and the International Nuclear Fuel-Cycle Evaluation (INFCE) Program which involves some 40 countries.

The specific goal of INFCE is to conduct technical and analytical studies of measures which can and should be taken at the national level and through international agreements to minimize the danger of the proliferation of nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes.

The Department of Energy requested that NWTS/ONWI support the U.S. efforts in the INFCE program, specifically to the INFCE Working Group 7: Waste Management and Disposal. As a contribution to this group, ONWI was requested as was its predecessor, OWI, to prepare various repository descriptions and to define the bases for those descriptions. Each repository description consisted of a definition of the representative site geology, the surface and subsurface engineered facilities, the operational procedures, and the types and amounts of wastes expected at the repository.

In its initial effort, May, 1978, OWI prepared repository descriptions for the U.S. representatives to the INFCE WG7. The descriptions considered four cases of repository design: LWR spent fuel in salt[1], LWR HLW in salt[2], LWR spent fuel in granite[3], and LWR HLW in granite[4]. These reports were presented to the INFCE WG7 meeting in Vienna, Austria, in May, 1978, as the U.S. position on repository concepts.

At the same time as this meeting, the Federal Republic of Germany (FRG)-Netherlands concept[5] was reviewed by the U.S., and a report on selected repository design issues[6] was prepared. This latter report discusses the advantages and disadvantages of the U.S. and FRG-Netherlands design bases with respect to the following topics:

- Domal versus bedded salt
- Shafts—number and utilization
- Multilevel construction
- Subterranean acreage requirements
- Multiple cans per hole; multiple assemblies per can
- Optional long-hole concept
- Bulk disposal
- Separation of waste and material handling
- Concern over flooding
- Thermal loading considerations
- Thermal criteria
- Critique of thermal analyses
- Critique of surface-uplift calculations
- Ventilation
- Design agreements.

As a result of the May meeting of the INFCE WG7, and in response to the comments on both the U.S. and FRG-Netherlands concepts, representatives of the FRG, the Netherlands, and the United States were directed to generate mutually acceptable bases for a description of a generalized reference repository in salt. From the design bases agreed upon by representatives of the three nations (which included thermal considerations, quantities of waste, waste-handling procedures, and ventilation procedures), designs for waste repositories in salt were generated.[7,8] These repository designs considered three more fuel-cycle cases than the May, 1978, descriptions: FBR with Pu-recycle, HWR once-through, and HWR with U and Pu-recycle.

These designs were subsequently modififed to reflect comments from other nations and two additional fuel-cycle cases: HWR with U and Th-recycle and HTR with UY and Th-recycle. Reports on these modified designs were generated in December, 1978[9,10], May, 1978[11,12] and August, 1979.[13]

The basic repository design was the now familiar room-and-pillar mined facility, such as that depicted in Figure 1. The designs incorporated appropriate facilities for handling and disposing of waste from the seven selected fuel cycles:

Cycle 1—LWR once-through
Cycle 2—LWR with U and Pu-recycle
Cycle 3—FBR with Pu-recycle
Cycle 4—HWR once-through
Cycle 5—HWR with U and Pu-recycle
Cycle 6—HWR with U and Th-recycle
Cycle 7—HTR with U and Th-recycle.

The site considerations for the INFCE salt design include: (1) the site is dry, that is, there are no substantial aquifers in the vicinity, (2) the host rock formation has sufficient thickness and lateral extent to provide space for the repository plus a sufficient margin of unmined rock, and (3) the host formation and surrounding strata are structurally intact and stable and should remain in this condition so that they can serve as a natural geologic containment barrier, preventing radionuclide migration into man's biosphere. The representative salt stratigraphy is presented in Figure 2. This is the contrast to the granite repository description being prepared by Sweden, in which the site contains sufficient amounts of groundwater that nuclide migration is of concern. For this situation, the Swedish design depends on engineered barriers to prevent nuclide migration, e.g., bentonite backfill, as the primary containment barrier.

The major controlling factors associated with the level of repository design appropriate for INFCE puposes are the acceptable repository areal thermal loadings and canister thermal loadings coupled with the total amount of waste arriving annually. Although the earlier designs were based on waste arrival rates and waste heat data generated largely from U.S. projections and ORIGEN code data, respectively, official INFCE waste arrival and heat data were used in the later designs.[14,15] On the basis of this waste-heat data, canister and repository loadings were calculated (Table 1). From these calculated loadings and the arrival rates, repository area requirements appropriate to support an annual nuclear power-generation rate of 100 GWe-yr were developed. Table 2 summarizes these requirements for waste emplaced in the repository at 10 years and 40 years after reactor discharge. The original repository concepts were based on a 10 year interim storage period prior to emplacement; however, the most current concepts considered a 40 year interim storage as well.

The repository areal thermal loadings were calculated in accordance with thermal criteria that were accepted for the INFCE repository designs. The acceptable thermal criteria were based on considerations agreed upon by the U.S., FRG, and the Netherlands. These thermal criteria may be summarized in three categories:

Very-near-field:

- Maximum HLW temperature of 500 C
- Maximum spent fuel-pin temperature of 200 C
- Only a limited volume of salt adjacent to the canister (such as 1 percent) shall be allowed to exceed 250 C.

Near-field:

- Only a certain percentage of the salt at the canister horizon (such as 25 percent) is allowed to exceed 200 C.

Far-field:

- A maximum surficial displacement due to thermal expansion and room closure of about 1.5 meters.

Foundation for these considerations may be found in the documentation of a study performed by Dr. J. E. Russell.[16]

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(1) Office of Waste Isolation, *U.S. Working Draft on Repository Physical Descriptions in a Salt Formation*, Union Carbide Corporation, Oak Ridge, Tennessee, Document number Y/OWI/SUB-78/22340/1, May 1978.

(2) Ibid.

(3) Office of Waste Isolation, *U.W. Working Draft on Repository Physical Descriptions in a Granite Formation*, Union Carbide Corporation, Oak Ridge, Tennessee, Document number Y/OWI/SUB-78/22340/2, May 1978.

(4) Ibid.

(5) Hamstra, J. and Velzeboer, P., *Design Study of a Radioactive Waste Repository to be Mined in a Medium Size Salt Dome*, Netherlands Energy Research Foundation, The Netherlands, ECN-78-023, January 1978.

(6) Science Applications, Inc., *Executive Style Briefings on Selected Repository Design Issues*, Performed under contract with the Office of Waste Isolation, Oak Ridge Tennessee, Document number Y/OWI/SUB-78-22340/4, June 2, 1978.

(7) U.S. Department of Energy, Appendix to the INFCE WG7 Report, *Technical Details of a Geologic Repository in Salt for the Disposal of Radioactive Waste*, Prepared by Science Applications, Inc., Oak Ridge, Tennessee, September 1978.

(8) U.S. Department of Energy, Text of Chapter III of the INFCE WG7 Report, *A Geologic Repository in a Salt Formation for the Disposal of Radioactive Wastes*, Prepared by Science Applications, Inc., Oak Ridge, Tennessee, September 1978.

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(10) U.S. Department of Energy, Text of Chapter III of the INFCE WG7 Report, *A Geologic Repository in a Salt Formation for the Disposal of Radioactive Wastes*, Prepared by Science Applications, Inc., Oak Ridge, Tennessee, December 1978.

(11) U.S. Department of Energy, Appendix to the INFCE WG7 Report, *Technical Details of a Geologic Repository in Salt for the Disposal of Radioactive Waste*, Prepared by Science Applications, Inc., Oak Ridge, Tennessee, May 1979.

(12) U.S. Department of Energy, Text of Chapter III of the INFCE WG7 Report, *A Geologic Repository in a Salt Formation for the Disposal of Radioactive Wastes*, Prepared by Science Applications, Inc., Oak Ridge, Tennessee, May 1979.

(13) U.S. Department of Energy, Appendix A to the INFCE WG7 Report, *Technical Details of a Geologic Repository in Salt for the Disposal of Radioactive Wastes*, Prepared by Science Applications, Inc., Oak Ridge, Tennessee, August 1979.

(14) INFCE Working Group 7, *Waste Management and Disposal for Selected Nuclear Fuel Cycle*, Draft Report, INFCE/WG7/26, May 1979.

(15) Heijboer, J. E., *Calculation of Waste Decay Heat for Seven Reference Fuel Cycles (INFCE Working Group 7)*, Energieonderzock Centrum Nederland, Petten, The Netherlands, April 1979.

(16) Russell, J. E., *Areal Thermal Loading Recommendations for Nuclear Waste Repositories in Salt*, Y-OWI/TM-37, Office of Waste Isolation, Oak Ridge, Tennessee, June 1979.

TABLE 1. THERMAL LOADINGS AND EMPLACEMENT DENSITIES

FUEL CYCLE	10 YR OLD WASTE			40 YR OLD WASTE		
	CANISTER THERMAL LOADING (kW/canister)	AREAL THERMAL LOADING (kW/hectare)	EMPLACEMENT DENSITY (canister/ha)	CANISTER THERMAL LOADING (kW/canister)	AREAL THERMAL LOADING (kW/hectare)	EMPLACEMENT DENSITY (canister/ha)
#1 LWR Once-Thru	.56	150	268	0.30	86	287
#2 LWR Pu-Recycle	1.86	370	199	0.74	270	365
#3 FBR Pu-Recycle	1.26	160	127	0.64	100	156
#4 HWR Once-Thru	.30	80	267	0.17	46	271
#5 HWR Pu-Recycle	2.29	250	109	0.95	140	147
#6 HWR U & Th-Recycle	1.56	350	224	0.77	350	455
#7 HTR U & Th-Recycle	1.09	320	294	0.53	290	547

TABLE 2. REPOSITORY AREA REQUIREMENTS TO SUPPORT 100 GWe-YR POWER CAPACITY

FUEL CYCLE	INDEPENDENT OF WASTE AGE		10 YR OLD WASTE[1]			40 YR OLD WASTE[1]		
	ANNUAL AREA FOR OTHER CANISTERED WASTE[2]	ANNUAL AREA FOR DRUMMED WASTE[3]	ANNUAL AREA FOR HLW	ANNUAL AREA FOR SPENT FUEL	TOTAL ANNUAL AREA[4]	ANNUAL AREA FOR HLW	ANNUAL AREA FOR SPENT FUEL	TOTAL ANNUAL AREA[4]
#1 LWR Once-Thru	0.2	12.1	-	28.0	40.3	-	26.1	38.4
#2 LWR Pu-Recycle	2.7	13.4	14.6	-	30.7	7.9	-	24.0
#3 FBR Pu-Recycle	6.0	6.6	18.1	-	30.7	14.7	-	27.3
#4 HWR Once-Thru	-	11.2	-	49.5	60.7	-	48.8	60.0
#5 HWR Pu-Recycle	2.0	15.3	26.6	-	43.9	19.7	-	37.0
#6 HWR U & Th-Recycle	3.1	18.6	13.4	-	35.1	6.6	-	28.3
#7 HTR U & Th-Recycle	0.5	25.8	9.5	-	35.8	5.1	-	31.4

[1] Time since discharge from reactor.

[2] This includes control rods, cladding, and reactor internals.

[3] This includes shielded and unshielded drums.

[4] Does not include initial development area of about 75 hectares (shafts, service areas, etc.) independent of power generation or fuel cycle.

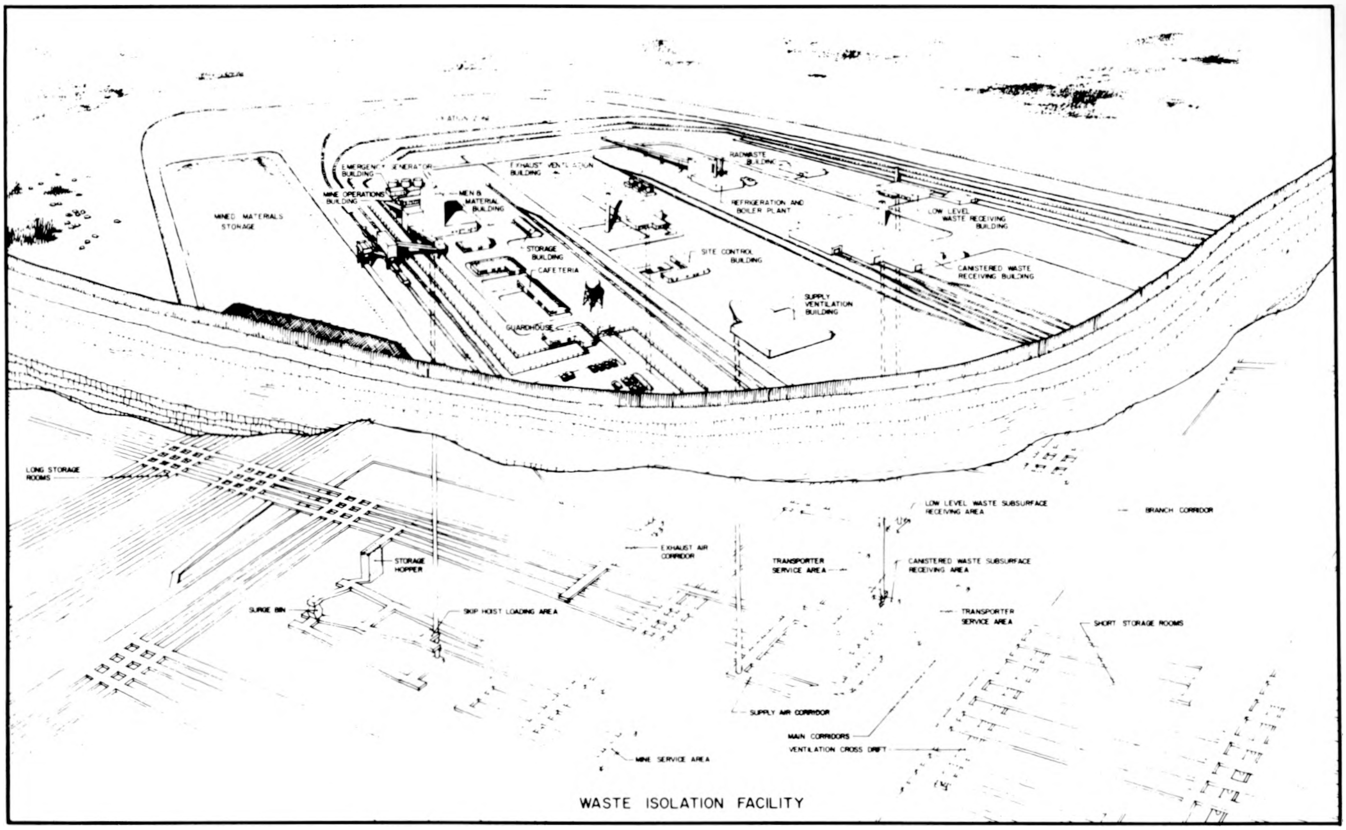

FIGURE 1. CUTAWAY VIEW OF WASTE REPOSITORY

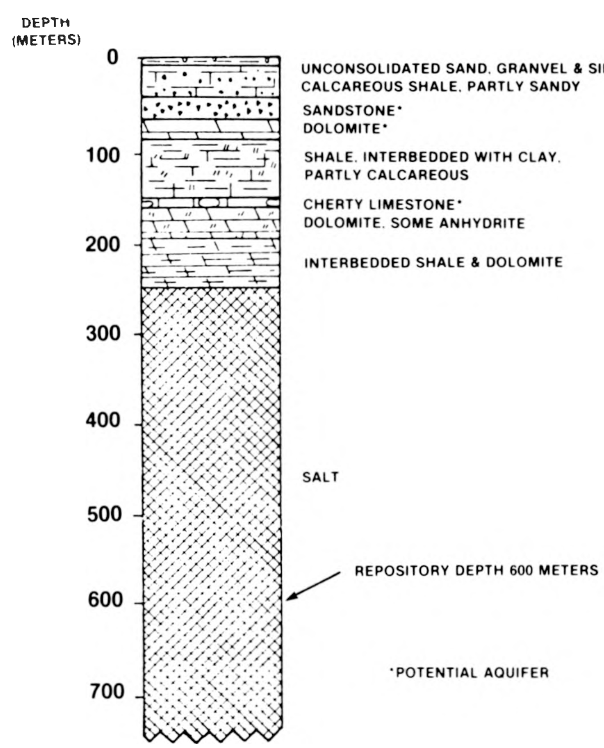

FIGURE 2. REPRESENTATIVE STRATIGRAPHIC SECTION FOR SALT