SANDIA REPORT

SAND97–2329 • UC–630
Unlimited Release
Printed September 1997

Preliminary Evaluation of the Use of the Greater Confinement Disposal Concept for the Disposal of Fernald 11e(2) Byproduct Material at the Nevada Test Site

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Distribution
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Preliminary Evaluation of the Use of the Greater Confinement Disposal Concept for the Disposal of Fernald 11e(2) Byproduct Material at the Nevada Test Site

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Abstract

This report documents a preliminary evaluation of the ability of the Greater Confinement Disposal boreholes at the Nevada Test Site to provide long-term isolation of radionuclides from the disposal of vitrified byproduct material.[d]

The byproduct material is essentially concentrated residue from processing uranium ore that contains a complex mixture of radionuclides, many of which are long-lived and present in concentrations greater than 100,000 picoCuries per gram. This material has been stored in three silos at the Fernald Environmental Management Project since the early 1950s and will be vitrified into 6,000 yd^3 (4,580 m^3) of glass "gems" prior to disposal.[e]

This report documents Sandia National Laboratories' preliminary evaluation for disposal of the byproduct material and includes: the selection of quantitative performance objectives; a conceptual model of the disposal system and the waste; results of the modeling; identified issues, and activities necessary to complete a full performance assessment.

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[d]The draft of this preliminary evaluation was completed in March of 1995 and this final version mirrors that original draft; subsequent actions and new information are provided in footnotes.
[e]Fernald is currently reevaluating this treatment option; some waste may be disposed as a cementitious material, rather than as glass gems.

Based on this preliminary evaluation: (1) the undisturbed wastes are likely to meet all regulatory requirements for the hypothetical member of the public; (2) the dose received by the hypothetical inadvertent intruder who disturbs the waste is not likely to exceed the acute dose standard; and (3) that intruder is likely to receive a chronic dose greater than the standard set by Chapter III of DOE Order 5820.2A.

This preliminary evaluation was completed in six weeks and does not provide enough information to make a definitive compliance decision. However, this evaluation provides a strong foundation for identifying and resolving outstanding issues necessary to make a compliance decision.

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List of Acronyms & Abbreviations

CFR	Code of Federal Regulations
CEDE	Committed Effective Dose Equivalent
DOE	U.S. Department of Energy
DOE/NV	Nevada Operations Office of the DOE
EDE	Effective Dose Equivalent
EPA	U.S. Environmental Protection Agency
FEMP	Fernald Environmental Management Project
GCD	Greater Confinement Disposal
ICRP	International Commission on Radiological Protection
l	liter
LLW	Low Level radioactive Waste
m	meter
mrem/yr	10^{-3} rem per year
MOP	Member of the Public
NEPA EIS	National Environmental Policy Act, Environmental Impact Statement
NTS	Nevada Test Site
OU4	FEMP Operable Unit 4 (OU4 includes the three silos)
Order 5820.2A	DOE Order 5820.2A, Radioactive Waste Management
pCi	pico Curies or 10^{-12} Curies
PA	Performance Assessment
QA	Quality Assurance
RCRA	Resource Conservation and Recovery Act
REECo	Reynolds Electrical and Engineering Company
RWMS	Radioactive Waste Management Site
Sandia	Sandia National Laboratories
TCLP	Toxic Characteristic Leaching Procedure
UIC	Underground Injection Control

Definitions

Accessible environment - The region at which contaminants are, or could be, in direct contact with humans as a function of everyday activities (e.g., at the ground surface, at a basement elevation, in the groundwater at a water supply well) (this is not "accessible environment" as defined in 40 CFR 191).

Byproduct - The tailings or waste produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content (from section 11e(2) of the Atomic Energy Act).

Committed effective dose equivalent (CEDE) - The sum of the committed dose equivalents to various tissues in the body, each multiplied by its weighting factor. It does not include contributions from external dose. Committed effective dose equivalent is expressed in units of rem (or sievert) (from DOE Order 5480.11).

Dose equivalent - The product of absorbed dose in rad (or gray) in tissue, a quality factor, and other modifying factors to account for differences in biological effectiveness due to the quality of radiation and its distribution in the body. Dose equivalent is expressed in units of rem (or sievert).

Effective dose equivalent (EDE) - The sum of the products of the absorbed dose and appropriate factors to account for differences in biological effectiveness due to the quality of radiation and its distribution in the body of reference man. The unit of the effective dose equivalent is the rem (from 40 CFR 61.91).

Inadvertent intruder* - A person who might occupy the disposal site after closure and engage in normal activities, such as agriculture, dwelling construction, or other pursuits in which the person might be unknowingly exposed to radiation from the waste (10 CFR 61).

Mixed Waste - Waste containing both radioactive and hazardous components as defined by the Atomic Energy Act and the Resource Conservation and Recovery Act, respectively. (from DOE Order 5820.2A)

Model - A conceptual description and the associated mathematical representation of a stated system, subsystem, component, or condition. A model describes the relationships between stated variables as a function of time, space, and initial or boundary conditions for a given purpose.

Performance Assessment - The quantitative process of evaluating the behavior of a disposal system and its components under a variety of expected and hypothetical conditions. Modeling is conducted to simulate the events and processes that might

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*This definition, from 10 CFR 61, is not the same as the definition used in the original draft report. This 10 CFR 61 definition better reflects the fact that direct contact with the waste is not necessary for an exposure.

affect the ability of the disposal system to limit releases to acceptable limits. A performance assessment is usually conducted to support decision making.

Special Case Wastes - (1) Those (radioactive) wastes which have limited or no planned disposal alternatives [from Wood, *et. al*, 1992].

Tailings - The remaining portion of a metal-bearing ore after some or all of such metal, such as uranium, has been extracted (from Uranium Mill Tailings Radiation Control Act of 1978).

Equation Symbols

C_i concentration of isotope i (mass or activity/unit volume of air, liquid or soil)
DCF_{iA} dose conversion factor for isotope i and pathway A (mrem/yr per unit concentration)
D_m molecular diffusion coefficient in water $[L^2/T]$
D_s effective gas diffusion coefficient in unsaturated sediments $[L^2/T]$
f_A a quality factor for the exposure pathway A
F radon flux $[activity/L^2T]$
k Emanation rate [activity or M/T]
K_d sorption coefficient $[L^3/M]$
M Atoms of Ra226 per unit mass of glass
m Activity of Ra226 per unit of glass
P Atoms of Th230 per unit mass of glass
p Activity of Th230 per unit of glass
S_e Bulk solubility of element e [mol/M]
S_i Solubility of isotope i [mol/M]
$t_{½}$ isotope half-life [T]
λ isotope decay constant $(\ln 2/t_{½})$ [1/T]
λ_M decay constant of ^{226}Ra [1/T]
λ_P decay constant of ^{230}Th [1/T]
θ volumetric water content [unitless]
τ tortuosity [L/L][unitless]

Executive Summary

The U.S. Department of Energy (DOE), Fernald Field Office, has requested approval to dispose of 11e(2) byproduct material at the Nevada Test Site (NTS). The majority of this byproduct material originated from the Shinkolobwe Mine in the Belgian Congo with the balance coming from the Rum Jungle Mine and Radium Hill Mine in Australia. This material has been stored in three silos at Fernald since the early 1950s and will be vitrified into 6,000 yd^3 (4,580 m^3) of glass "gems" prior to disposal.[†]

To determine if the NTS is an appropriate location for this material, the Nevada Operations Office of the U.S. Department of Energy (DOE/NV) is gathering additional information. Since a significant portion of the radioactive inventory in the byproduct material is composed of long-lived radionuclides, DOE/NV is evaluating the possibility of utilizing the Greater Confinement Disposal (GCD) concept. In January 1995, Sandia National Laboratories (Sandia) was asked to evaluate the possibility of using the GCD concept for the disposal of this byproduct material.

The GCD concept consists of large diameter boreholes, approximately 10 ft (3 m) in diameter and 120 ft (37 m) deep. Coupled with a very arid environment and a thick unsaturated zone (groundwater is 650 ft (198 m) below the bottom of the boreholes) the GCD concept appears to provide a practical method for the disposal of special-case transuranic wastes requiring 10,000 years of isolation (Price et al., 1993).

This report documents Sandia's preliminary evaluation for 11e(2) waste disposal and includes: 1. approach for providing a preliminary evaluation; 2. regulatory analysis; 3. selection of performance objectives; 4. site and waste description; 5. results of modeling; 6. identification of issues; 7. assumptions; and 8. activities necessary to complete a full Performance Assessment (PA).

Based on this preliminary evaluation: (1) the undisturbed wastes are likely to meet all regulatory requirements for the hypothetical member of the public; (2) the dose received by the hypothetical inadvertent intruder who disturbs the waste is not likely to exceed the acute dose standard; and (3) that intruder is likely to receive a chronic dose greater than the standard set by DOE Order 5820.2A. It is important to note that this hypothetical intruder would receive an unacceptable chronic dose independent of the site setting because it is primarily a function of the source (i.e., burial at NTS and burial in Ohio will provide approximately the same results).

Because a number of issues remain unresolved, this preliminary evaluation does not provide enough information to make a definitive disposal decision. However, this evaluation provides a strong foundation for identifying and resolving outstanding issues that need to be addressed in order to make a disposal decision.

[†]Fernald is currently reevaluating this treatment option; some waste may be disposed as a cementituous material, rather than as glass gems.

1.0 Introduction

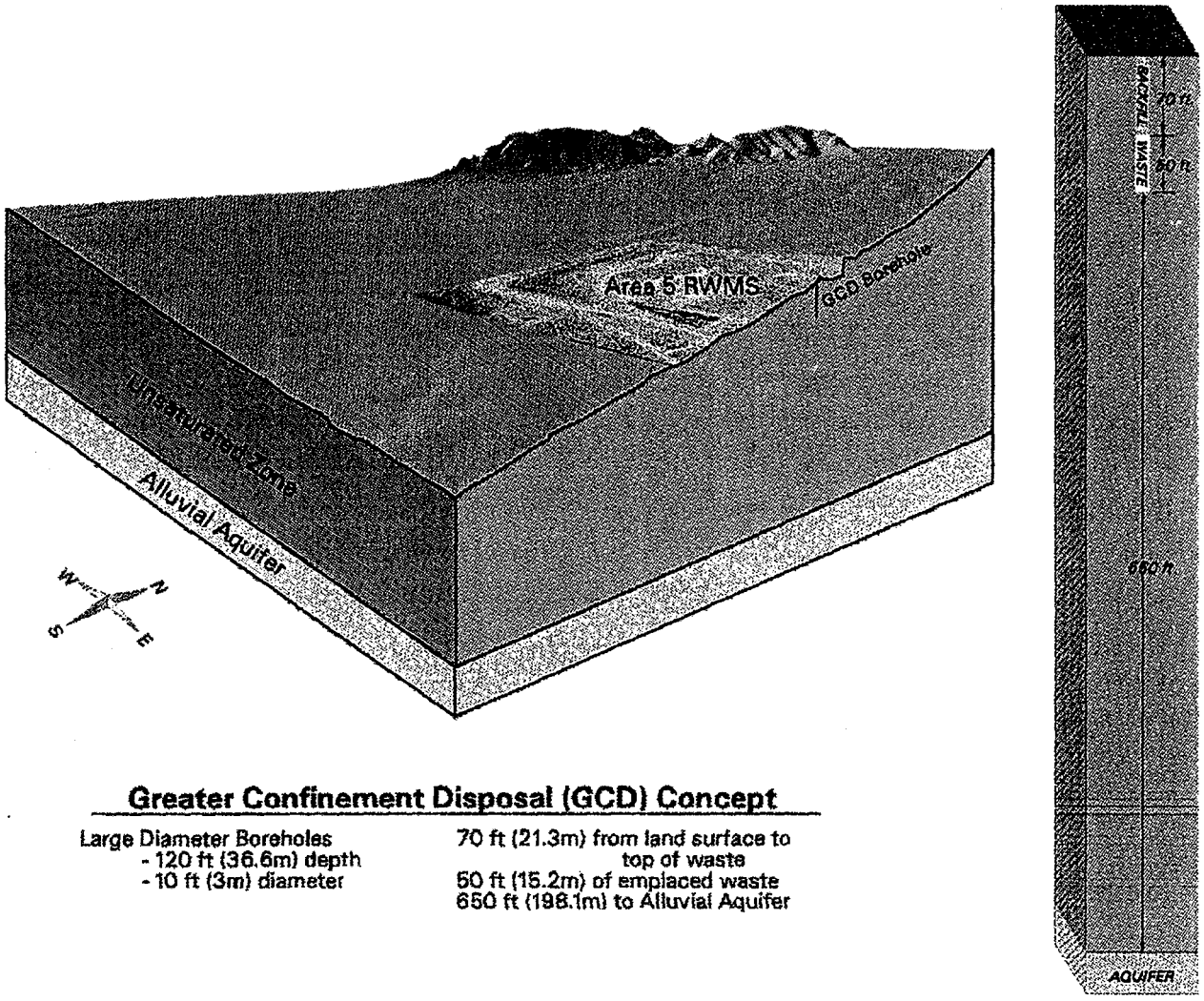
This report documents the preliminary evaluation of the suitability of using Greater Confinement Disposal (GCD) boreholes for the disposal of certain radioactive wastes at the Nevada Test Site (NTS).

The U.S. Department of Energy (DOE), Fernald Field Office, has requested approval to dispose of 11e(2) byproduct material at the NTS. To determine if the NTS is an appropriate location for this material, DOE, Nevada Operations Office (DOE/NV) is gathering additional information. In a January, 1995, memorandum, DOE/NV asked Sandia National Laboratories (Sandia) to evaluate the possibility of using the GCD concept for the disposal of this byproduct material. Sandia is currently conducting a Performance Assessment (PA) to evaluate the ability of the GCD facility to isolate transuranic (TRU) radioactive wastes as required by 40 CFR 191.

The byproduct material is essentially concentrated residue from processing uranium ore that contains a complex mixture of radionuclides, many of which are long-lived and present in concentrations greater than 100,000 picoCuries per gram (see Appendix 1). The majority of this byproduct material originated from the Shinkolobwe Mine in the Belgian Congo with the balance coming from the Rum Jungle Mine and Radium Hill Mine in Australia. Prior to disposal, candidate byproduct materials will be vitrified into 6,000 yd^3 (4,580 m^3) of glass "gems." Since a significant portion of the radioactive inventory in the glass gems is composed of long-lived radionuclides, DOE/NV is evaluating the possibility of utilizing the GCD boreholes.

The GCD concept is a practical method for intermediate-depth burial of wastes. As currently used at the NTS, the GCD concept consists of large diameter boreholes, approximately 10 ft (3 m) in diameter and 120 ft (37 m) deep (see Figure 1). The lower 50 ft (15.2 m) of a borehole is filled with wastes, and the upper 70 ft (21.3 m) consists of native backfill. Coupled with a very arid environment and a thick unsaturated zone (groundwater is 650 ft (198 m) below the bottom of the boreholes), the GCD concept appears appropriate for the disposal of special-case TRU wastes which require 10,000 years of isolation (Price et al., 1993). The GCD concept has also been proposed for the disposal of other special case, or orphan, radioactive wastes (see Attachment A).

This report documents Sandia's preliminary evaluation and includes: 1. approach for providing a preliminary evaluation; 2. regulatory analysis; 3. selection of performance objectives; 4. site and waste description; 5. results of modeling; 6. identification of issues; 7. assumptions; and 8. activities necessary to complete a full PA.



Greater Confinement Disposal (GCD) Concept

Large Diameter Boreholes
- 120 ft (36.6m) depth
- 10 ft (3m) diameter

70 ft (21.3m) from land surface to top of waste
50 ft (15.2m) of emplaced waste
650 ft (198.1m) to Alluvial Aquifer

Figure 1. Schematic of GCD concept.

1.1 Scope of Work

Sandia, in a letter from DOE/NV dated January 12, 1995 (Attachment B), was asked to provide "... a determination as to whether this Fernald Environmental Management Project (FEMP) waste could be disposed of in the GCD units, and if necessary, what issues need to be resolved." From this letter, the following Scope of Work was derived:

1. Determine if the vitrified byproduct wastes can be disposed of in GCD units;

2. Identify the issues that need to be resolved, specifically those that affect waste management activities (e.g., monitoring, permitting, Underground Injection Control (UIC) and closure);

3. Identify options for unresolved issues, and

4. Complete the assessment by March 1, 1995.

This preliminary evaluation concentrates on waste disposal, not waste operations. Operational issues, such as occupational radiological doses to workers at the Area 5 Radioactive Waste Management Site (RWMS) were not addressed in this report, except to calculate radon flux at the surface. The current operating contractor, Reynolds Electrical and Engineering Company (REECo)[1] would be better prepared to answer these operational questions. In Chapter 8.0, this report does identify (and offer suggestions on) a number of issues relative to waste disposal.

Wendy Griffin, DOE/NV, agreed to extend the deadline for this evaluation until March 15, 1995 and Joe Ginanni, DOE/NV, has agreed to the transfer of $60,000 from the existing Sandia (40 CFR 191) PA work to fund this preliminary evaluation.

1.2 Approach

The following approach was used to make a preliminary evaluation of the acceptability of the GCD concept for the disposal of these Fernald wastes.

1. Determine which regulation(s) govern the disposal of the vitrified byproduct materials using the GCD concept at the NTS, then;

2. Determine the quantitative measures (performance objectives) that are defined by the regulations(s), and

3. Evaluate the disposal configuration against the performance objectives. This step includes defining the conceptual model, determining the appropriate numerical models and model input parameter values, and using the model results to evaluate the sites performance.

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[1]Bechtel Nevada has replaced REECo as the operating contractor.

Many of the steps in this approach were abbreviated so that a preliminary evaluation could be completed in a short time period.

2.0 Regulatory Analysis

The first step in our evaluation was to determine the requirements that are, or may be, applicable to the disposal of Fernald byproduct waste. The material to be disposed of consists of waste produced by the extraction or concentration of uranium or thorium from ore processed primarily for its source material content, and is legally classified as "byproduct" as defined in section 11e(2) of the Atomic Energy Act.

Byproduct material is regulated by the DOE. DOE's regulations for the disposal of byproduct material are contained in Chapter IV of DOE Order 5820.2A (Radioactive Waste Management, 9/26/88). Chapter IV allows "small volumes" of byproduct material to be disposed of "... in accordance with the requirements of Chapter III. The amount of waste that Fernald wishes to dispose of could be interpreted as a small volume,[2] and Fernald would like to dispose of this byproduct material under the requirements of Chapter III.

Therefore, this regulatory analysis is based on the requirements set forth in Chapter III of DOE Order 5820.2A (Order 5820.2A). Chapter III sets requirements for the disposal of DOE's low level radioactive waste (LLW).

Paragraph 3(i)1 of Chapter III of Order 5820.2A states that:

> Low-level waste shall be disposed of by methods appropriate to achieve the performance objectives stated in paragraph 3(a), consistent with the disposal site radiological performance assessment in paragraph 3(b).

Paragraph 3(a) of Chapter III states that DOE LLW shall be disposed in a manner that meets the four following criteria. These four criteria are the foundation for our performance objectives.

> 1. Protect public health and safety in accordance with standards specified in applicable EH Orders and other DOE Orders.
>
> 2. Assure that external exposure to the waste and concentrations of radioactive material which may be released into surface water, ground water, soil, plants, and animals results in an effective dose equivalent that does not exceed **25 mrem/yr to any member of the public**. Releases to the atmosphere shall meet the requirements of 40 CFR 61. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable.
>
> 3. Assure that the committed effective dose equivalents received by individuals who inadvertently may intrude into the facility after the loss of

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[2]The 6,000 yd^3 is a small volume compared to the volume of waste which is disposed annually at the NTS.

active institutional control (100 years) will not exceed **100 mrem/yr for continuous exposure or 500 mrem for a single acute exposure**.

4. Protect ground water resources, consistent with Federal, State, and local requirements. (emphasis added)

Paragraph 3(b) of Chapter III of Order 5820.2A states that a PA shall be prepared and maintained for the purposes of demonstrating compliance with the four criteria of paragraph 3(a). Therefore, demonstrating compliance with paragraph 3(a) will require an approved PA. This report is not a PA; however, this preliminary evaluation provides a strong foundation for a PA.

Paragraph 3(a)1 of Chapter III includes by reference, the standards contained in DOE Order 5400.5. Chapter II of DOE Order 5400.5 (Radiation Protection of the Public and the Environment, 2/8/90) specifies doses (100 mrem/yr EDE from all pathways, 10 mrem/yr EDE air pathway, and 4 mrem/yr EDE from drinking water) and radionuclide concentrations in groundwater (5 pCi/l for ^{226}Ra and ^{228}Ra, 15 pCi/l for gross alpha including ^{226}Ra but excluding Rn and U) as a means of protecting members of the public from radiation. The issue of point of compliance is relevant here, as the 100 mrem/yr EDE (all pathways) and the 10 mrem/yr EDE (air) apply to members of the public, while the 4 mrem/yr and radionuclide concentration requirements apply to a public drinking water supply operated by the DOE, the location of which is not specified. These standards apply to all DOE operations.

Paragraph 3(a)1 of Chapter III also includes, by reference, DOE Order 5400.1 (General Environmental Protection Program), which includes, by reference, a number of environmental protection standards that are mandatory for all DOE operations, to the extent legally applicable. On this list are 12 Executive Orders, covering such topics as "Exotic Organisms," "Protection and Enhancement of Cultural Environment," and "Off-Road Vehicles on Public Lands." These environmental protection standards have not been researched, but they probably do not contain quantitative requirements that can be incorporated in a performance evaluation. The list also documents various Federal Acts that are mandatory for all DOE Operations. These are: Protection of Historic and Cultural Properties; Protection of Archaeological Resources; Clean Air Act; Clean Water Act; Safe Drinking Water Act; Coastal Zone Management Act of 1972, As Amended; various radiation protection standards (40 CFR 190 - 192); CERCLA; Federal Insecticide, Fungicide, and Rodenticide Act, as amended; Resource Conservation and Recovery Act (RCRA); Endangered Species Act of 1973, as amended; Toxic Substances Control Act; Noise Control Act; and the Wilderness Act, as amended.

Paragraph 3(a)2 of Chapter III of Order 5820.2A establishes a 25 mrem/yr effective dose equivalent (EDE) requirement for members of the public (MOP). The point of compliance, the point at which one determines if the 25 mrem/yr dose is exceeded, is not defined in the Order. This 25 mrem/yr is the sum of the dose from the following pathways: surface water; groundwater; soil; plants; and animals and does not seem to include the air pathway, which is addressed separately.

Paragraph 3(a)2 also sets a 10 mrem/yr EDE standard for air dose (excluding radon) to a MOP and an emanation rate of less than 20 $pCi/m^2/s$ for ^{222}Rn (as specified by 40 CFR 61.92 and 61.192 as part of the Clean Air Act). The 10 mrem/yr EDE requirement applies to a MOP, although the point of compliance and the time frame are not defined. The 20 $pCi/m^2/s$ emanation rate applies in the air as an average for the entire source, where the source is defined to be the area used for disposal. This requirement is consistent with 40 CFR 192.32, the Environmental Protection Agency's (EPA's) requirements for management of uranium mill tailings.

In paragraph 3(a)3 of Chapter III of Order 5820.2A, the requirements for protection of persons who may inadvertently intrude into the wastes, after institutional control is lost, is in terms of a committed effective dose equivalent (CEDE), rather than EDE, the difference being that CEDE does not include the dose from external exposure. Additionally, the circumstances surrounding the inadvertent intrusion (e.g., well drilling) are not defined in the Order.

Chapter IV of Order 5820.2A states that it is DOE's policy to dispose of byproduct material "consistent with the requirements of the residual radioactive material guidelines contained in 40 CFR 192." However, 40 CFR 192 may not be applicable as these wastes are going to be managed under Chapter III of Order 5820.2A and not under Chapter IV of the Order. It probably does not make much difference since the 20 $pCi/m^2/s$ for ^{222}Rn (in 40 CFR 192) is also required by other regulations.

The requirements in 40 CFR 141.15 and 141.16 are included here because of the requirement of paragraph 3(a)4 to "protect ground water resources, consistent with Federal, State, and local requirements." These requirements are part of the Safe Drinking Water Act and are identical to the groundwater protection requirements in DOE Order 5400.5, with three exceptions. First, in 40 CFR 141.16, the 4 mrem/yr dose requirement is in terms of dose equivalent, not effective dose equivalent. Second, the 4 mrem/yr dose is from man-made beta- and photon-emitting radionuclides, not from all radionuclides. Third, the requirement is not restricted only to DOE-operated water supplies. It applies to any water supply.

An issue relevant to paragraph 3(a) of Chapter III of Order 5820.2A is the time of compliance. The time frame in which one calculates potential doses can be divided into three segments: operational, institutional control, and post closure. During the time in which the facility is actively disposing of wastes, the facility is operational. For 100 years after disposal operations cease, the facility is, by definition, under institutional control. The third time period, post closure, begins 100 years after disposal operations cease. With the exception of the intruder protection requirements, Order 5820.2A does not specify whether the requirements of Order 5820.2A apply during the operational phase of the disposal site, during the institutional control phase, during the post-closure phase, or during all three. Order 5820.2A also does not specify the length of the post-closure period for which the site must meet the requirements.

Table 1 summarizes the requirements we have extracted from the Order.

In addition to those regulations set in paragraph 3(a) of Chapter III, which define standards for the disposal of LLW, there are a number of other regulatory issues which may need to be considered. These include compliance with the National Environmental Policy Act (NEPA), and the potential applicability of Nevada's Mixed Waste Authority and Nevada's UIC program.

If the current site-wide NEPA Environmental Impact Statement (EIS) for the NTS is sufficiently broad as to cover the disposal of this waste using the GCD concept, no significant NEPA actions would be required. If the site-wide EIS does not cover the use of the GCD disposal concept for these wastes, a separate NEPA action will be necessary. Such action would probably involve the development and approval of a separate NEPA Environmental Assessment or EIS. Sufficient lead time and funds will need to be allocated for the completion of the NEPA process.

On June 29th, 1992 the U.S. EPA granted Nevada authority to regulate mixed wastes, that is, wastes containing both radioactive and hazardous components as defined by the Atomic Energy Act and the RCRA, respectively. The State may want to use this authority to regulate the disposal of these silo wastes. Prior to vitrification, these wastes met the definition of a RCRA hazardous waste because they failed the EPA's Toxicity Characteristics Leaching Procedure (TCLP) test. However, the vitrified wastes pass the TCLP test and no longer qualify as a RCRA hazardous waste (Operable Unit 4 Treatability Study Report ..., 1993). Additionally, byproduct material is specifically excluded from RCRA regulations by the Byproduct Rule (10 CFR 962).

The most important of the State's concerns may involve the interpretation of underground injection. The State has taken the position that the GCD Boreholes are "Class IV" injection wells as defined under the Safe Drinking Water Act regulations of 40 CFR 144.13. Interpretation of the GCD boreholes as injection wells would prevent the future use of the boreholes for waste disposal; such an interpretation might also be applied to geologic repositories for high level radioactive wastes.

On the Federal side, in the 1993 version of 40 CFR 191 (58 FR 66407), the EPA has taken the position that the disposal of 40 CFR 191 wastes (e.g., TRU wastes) does not constitute underground injection. This EPA position may provide an important basis for countering the States definition of underground injection (since the UIC regulations originated with the EPA). We believe that the GCD concept does not constitute underground injection as envisioned by the UIC regulations; however, this issue still must be resolved with the State.

An alternative that avoids the injection well issue would be to bury the Fernald wastes in a very deep trench, say 37 m deep and 150 m wide with 21 m of backfill. Such a trench is a different version of the GCD disposal concept. A trench 150 m wide and 37 m deep would not require special equipment or expertise to construct, and would allow a high degree of control over waste placement and backfill compaction (Price, 1994 provides a discussion of boreholes vs. trenches). Although we do not favor the

Table 1. Requirements Derived from DOE Order 5820.2A

Regulation	Exposure Group	Exposure Mode	Performance Objective	Compliance Point	Period of Compliance
DOE Order 5820.2A Ch. III	MOP[1]	surface water, groundwater, soil, plants, animals, excludes air(?)	25 mrem/yr EDE[2]	not defined	not defined
40 CFR 61.92 (incorporated by reference)	MOP	air	10 mrem/yr EDE excluding radon	not defined	not defined
40 CFR 61.192 (incorporated by reference)	NA	NA	20 pCi/m^2/s ^{222}Rn	In air directly above area used for disposal	not defined
DOE Order 5820.2A Ch. III	Intruder	not defined	100 mrem/yr CEDE[3] (chronic exposure)	not defined	Post-closure, beginning 100 years after closure, end point not defined
DOE Order 5820.2A Ch. III	Intruder	not defined	500 mrem CEDE (single acute exposure)	not defined	Post-closure, beginning 100 years after closure, end point not defined
DOE Order 5400.5 (incorporated by reference)	MOP	all	100 mrem/yr EDE	Point of maximum exposure	during DOE operations
DOE Order 5400.5 (incorporated by reference)	MOP	air	10 mrem/yr EDE	Point of maximum exposure	during DOE operations
DOE Order 5400.5 (incorporated by reference)	Persons consuming water	drinking water	4 mrem/yr EDE	Public drinking water supply operated by DOE	during DOE operations
DOE Order 5400.5 (incorporated by reference)	NA	NA	5 pCi/l ^{226}Ra and ^{228}Ra	Public drinking water supply operated by DOE	during DOE operations

Table 1. Requirements Derived from DOE Order 5820.2A (Continued)

Regulation	Exposure Group	Exposure Mode	Performance Objective	Compliance Point	Period of Compliance
DOE Order 5400.5 (incorporated by reference)	NA	NA	15 pCi/l gross alpha, including ^{226}Ra, excluding Rn and U	Public water supply operated by DOE	during DOE operations
40 CFR 141.16 (incorporated by reference)	Persons consuming water	drinking water	4 mrem/yr man-made beta and photon emitters	At the tap of any user	ongoing
40 CFR 141.15 (incorporated by reference)	NA	NA	5 pCi/l ^{226}Ra and ^{228}Ra	At the tap of any user	ongoing
40 CFR 141.15 (incorporated by reference)	NA	NA	15 pCi/l gross alpha, including ^{226}Ra, excluding Rn and U	At the tap of any user	ongoing
40 CFR 192.02	NA	NA	20 pCi/m^2/s ^{222}Rn	In air directly above area used for disposal	ongoing
40 CFR 192.02	NA	NA	0.5 pCi/l increase in ^{222}Rn	In air at any location outside the disposal site	ongoing

1. MOP - member of the public 2. EDE - Effective dose equivalent 3. CEDE - Committed effective dose equivalent
DOE Order 5820.2A - Radioactive Waste Management
DOE Order 5400.5 - General Environmental Protection Program
40 CFR 61 - National Emissions Standards for Hazardous Air Pollutants
40 CFR 141 - National Interim Primary Drinking Water Regulations
40 CFR 192 - Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings

excavation of such a trench, due the increased operational safety hazard and because the construction of deep trenches is very inefficient, it would avoid the State's concern (and possibly a lawsuit) that the GCD boreholes constitute underground injection.

3.0 Performance Objectives

There is no single performance objective, or standard, for determining if a particular disposal option is appropriate for a particular waste. A standard which is appropriate for LLW may be inappropriate for a hazardous chemical waste. This section defines, and defends, what are believed to be appropriate performance objectives for the preliminary evaluation of the use of the GCD boreholes for the disposal of the vitrified Fernald byproduct materials.

As discussed above, the regulation which specifically governs these byproduct materials is DOE Order 5820.2A (Order 5820.2A), Chapter IV. Chapter IV allows small quantities of byproduct material to be disposed of as LLW. There is a small quantity of these byproduct materials and FEMP desires to dispose of them as LLW. Therefore, these wastes must be disposed of in a manner that meets the performance objectives for the disposal of LLW, which are Chapter III of the Order.

Chapter III of Order 5820.2A incorporates, by reference, a number of other standards and also establishes three standards. The three standards specific to Order 5820.2A are a maximum dose (25 mrem/yr all pathways, except air) to the general public, a maximum chronic dose (100 mrem/yr continuous exposure) to an inadvertent intruder, and a maximum acute dose (500 mrem single exposure) to an inadvertent intruder. These dose limits are conservative; the "background" dose to an individual in the U.S. is in the 300 to 400 mrem/yr range (Wood et al., 1992, p. 31).

As discussed in Section 2 many important performance objectives are not defined in the Order. This problem was recognized and DOE/HQ/EM-35 established a Performance Assessment Task Team to integrate the activities of the DOE facilities that are preparing PAs for the disposal of new LLW as required by Chapter III of the Order. The intent of the PA Task Team is to achieve a degree of consistency among PAs and to recommend policy and guidance to DOE on issues that impact the PA including release scenarios and parameters (Wood et al., 1992 and Wood et al., 1994).

Essentially, Order 5820.2A requires the evaluation of the performance under two distinct and separate sets of circumstances. The first set of circumstances (known as a scenario), involves protecting a hypothetical MOP from the buried wastes assuming that the wastes remain undisturbed. The second scenario involves protecting a hypothetical intruder who physically disturbs the buried wastes.[3] In both cases knowledge of the site's history is lost. Order 5820.2A provides little guidance concerning either of these scenarios.

The quantitative requirements from DOE Orders 5400.1 and 5400.5 are not used as performance objectives, because both of these Orders are only applicable to "DOE

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[3]The inadvertent human intruder could receive a dose by occupying the disposal site, without disturbing the waste. However, because the waste will be 70 feet deep, the bounding scenario requires the intruder to physically disturb the waste.

operations." By definition, our evaluation commences with the loss of institutional control (i.e., post closure and after the NTS ceases to be a DOE operation).

For the protection of groundwater, 3(a)4 of Chapter III of Order 5820.2A requires compliance with standards that are "consistent with Federal, State, and local requirements." Therefore, we will retain the requirements of the Safe Drinking Water Act, 40 CFR 141.

Table 2 summarizes the performance objectives used in this preliminary evaluation. The performance objectives selected for this evaluation are based on the Order, recommendations by Wood et al. (1992, 1994) and our interpretation of the specifics of this evaluation. The metrics listed in Table 2 are discussed in more detail in the following sections.

3.1 Inadvertent Intruder Scenario

Order 5820.2A states that the inadvertent intruder cannot receive more than 100 mrem CEDE chronic dose, and no more than 500 mrem CEDE[4] acute dose from all pathways, including radon. Order 5820.2A provides no guidance concerning the inadvertent intruder scenario, except that the intruder standard applies after loss of institutional control.

The probability that a person would inadvertently disturb wastes buried 21 m (70 ft) beneath the surface of a very remote section on Nevada is thought to be very low. However, Order 5820.2A establishes a dose standard for the inadvertent intruder and a scenario must be established and evaluated to determine if this standard is met (i.e., probability of occurrence is not addressed).

Therefore, we must define a scenario involving a hypothetical person who physically disturbs the buried wastes. Wood et al. (1994) recommends analysis of facility performance under the following inadvertent intruder scenarios:

1. An acute construction scenario and a chronic agricultural (homesteader) scenario involving excavating into disposal units, mixing exhumed waste in an intruder's vegetable garden, and permanent residence in a home on top of disposal units;

2. An acute discovery scenario and a chronic residential scenario involving an attempted excavation into disposal units, which is assumed to be precluded by the presence of intact engineered barriers, and

3. An acute drilling scenario and a chronic post-drilling scenario involving drilling through disposal units and mixing the drilling wastes in an intruder's vegetable garden.

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[4]Doses to the inadvertent human intruder were calculated as EDE as per the Wood, et al. (1994) guidance.

Table 2. Performance Objectives

Regulation	Exposure Group	Pathway	Performance Objective	Compliance Point	Scenario
DOE Order 5820.2A Ch. III	MOP	surface water, ground water, soil, plants, animals, excludes air	25 mrem/yr EDE	on site	Residential, undisturbed
40 CFR 61.92 (incorporated by reference)	MOP	air	10 mrem/yr EDE excluding radon	on site	Residential, undisturbed
40 CFR 61.192 (incorporated by reference)	NA	air	20 pCi/m^2/s ^{222}Rn	on site, in air directly above disposal area	undisturbed
DOE Order 5820.2A Ch. III	Intruder	surface water, groundwater, soil, plants, animals, air	100 mrem/yr EDE (chronic exposure)	on site	Homesteader, intruding
DOE Order 5820.2A Ch. III	Intruder	surface water, groundwater, soil, plants, animals, air	500 mrem EDE (single acute exposure)	on site	Homesteader, intruding
40 CFR 141.16 (incorporated by reference)1	persons consuming the water	drinking water	4 mrem/yr DE man-made beta and photon emitters	at the tap of any user	undisturbed
40 CFR 141.16 (incorporated by reference)	NA	NA	5 pCi/l ^{226}Ra and ^{228}Ra	at the tap of any user	undisturbed
40 CFR 141.16 (incorporated by reference)	NA	NA	15 pCi/l gross alpha	at the tap of any user	undisturbed

Of these three recommended scenarios, the concept of homesteader settling on the Area 5 RWMS and drilling a water well through a GCD borehole is the most plausible. Spreading the drill cuttings from the water well in the vegetable garden was deemed unreasonable and is not assessed. Intuitively, the drill cuttings would not be spread in the vegetable garden because glass shards, metal and concrete fragments would make poor soil. It is not physically reasonable to assume that an inadvertent intruder would excavate wastes that are buried 21 m deep, and therefore the second suggested scenario is not considered. Consequently, we define the following acute and chronic inadvertent intruder scenario:

> Sometime after the loss of institutional control, a hypothetical individual, a homesteader, moves on to the Area 5 RWMS and builds a house, with a basement, over one of the GCD boreholes that contains Fernald byproduct material. This homesteader drills a water well directly through another of the Fernald GCD boreholes. Finally, the homesteader grows a vegetable garden over a third Fernald GCD borehole. This hypothetical homesteader is evaluated against the acute and chronic standards set for an intruder. Specific details (e.g., the number of pounds of vegetables eaten per year) are discussed in Section 5.

3.2 Member of Public Scenario

Order 5820.2A states that the MOP should not receive more than 25 mrem per year (all pathways, except air) and 10 mrem per year from the air pathway (excluding radon). Order 5820.2A provides no guidance concerning the MOP scenario.

Wood et al. (1994) develop the concept of a buffer zone surrounding the disposal site and assume that the MOP would remain outside that boundary. With this recommendation, the point of compliance would be the point of maximum exposure beyond the buffer zone.

We assume that institutional knowledge of the site is lost, and there is little reason for a MOP to remain beyond some unmarked boundary.[5] Therefore, we define the following MOP scenario.

> Sometime after loss of institutional control, a hypothetical MOP moves on site where the GCD boreholes are located. This MOP drills a water well in an area away from the GCD boreholes and builds a house (without a basement) over one of the boreholes. The individual then lives on site, drinking water from the clean well and eating food from offsite. This hypothetical MOP is evaluated against the MOP standard. Specific details (e.g., the number of hours spent outside) are discussed in Section 5.

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[5]The standard protocol assumes the MOP stays 100 m away and the intruder occupies the disposal site. With wastes buried at 70 ft deep, placing the MOP on the waste site is not a critical error as it bounds the MOP dose estimate.

3.3 Radon Flux

Three quantitative performance objectives are defined for the release of radon. First, as a stand-alone standard, the average flux of radon at the land surface should not exceed 20 $pCi/m^2/s$. The dose from this radon flux is not added to the dose received by the MOP. Second, the inadvertent intruder cannot receive more than 500 mrem (CEDE), acute dose, including radon, and third, the inadvertent intruder cannot receive more than 100 mrem per year (CEDE) chronic dose, including radon.

3.4 Time of Compliance

Regulations for the safe disposal of radioactive waste typically include the time frame of compliance. For example, caps constructed over uranium mill tailings shall be designed to be effective for up to 1,000 years, to the extent reasonably achievable, and, in any case, for at least 200 years (40 CFR 192.02(a)). Order 5820.2A implies compliance for all future time.

Wood et al. (1994) discuss this issue at some length and recommend using 10,000 years as the period of compliance. They suggest that if peak dose does not occur within 10,000 years, modeling should continue beyond that time, although more as a mathematical exercise rather than as a compliance evaluation. In order to evaluate peak doses, we have not limited the time of compliance.

3.5 Calculation of Dose

Order 5820.2A establishes performance objective in terms of CEDE as well as EDE. A CEDE does not include contributions from external dose. For this preliminary evaluation, all dose is assessed as EDE and is calculated using the GENII-S computer code (Napier et al., 1988), which uses ICRP 30 and 26 models for all isotopes and exposure pathways other than radon inhalation. The dose to the intruder due to inhalation of radon is estimated using the same empirical models that were used in the Savannah River PA for the E-Area Vaults disposal facility (Martin Marietta Energy Systems et al., 1994) and the second PA for the GCD site (Baer et al., 1994). The dose simulation models are discussed in greater detail in Section 5.3.

4.0 Disposal Facility Description

4.1 Disposal Site Characteristics

The NTS is a DOE facility occupying approximately 1,350 mile^2 (3,500 km^2) in southeastern Nevada, approximately 65 miles (105 km) northwest of Las Vegas. Activities at the NTS include a variety of nuclear and non-nuclear projects and experiments as well as waste management operations. One of these waste management operations is the Area 5 RWMS, which encompasses 732 acres (296 hectares) at an elevation of 3200 ft (975 m) in the southeastern section of the NTS (See Figure 2).

The Area 5 RWMS was established in 1978 for the purpose of disposing of LLW generated at the NTS and other DOE facilities. Current waste disposal operations at the Area 5 RWMS include disposal of classified and unclassified LLW in trenches and pits. The Area 5 RWMS also contains 13 GCD boreholes, which are described in the next section.

The Area 5 RWMS is situated near the edge of an intermontane basin and is underlain by unconsolidated deposits of cobbles, gravel, sand, and silt resulting from the weathering of nearby mountains. The climate at the Area 5 RWMS is typical of the upper Mojave Desert, with average daily temperatures ranging from 2°C in January to 24°C in August. Precipitation typically occurs as isolated, short duration, intense summer thunderstorms and regional, long duration, low intensity winter storms. Average annual precipitation in 5 in. (12.5 cm) and estimated evaporation potential is 71 in. (181 cm) (French, 1993).

In this basin the water table is up to 780 feet (240 m) below the land surface. The thick unsaturated zone, coupled with climate change, has resulted in a situation in which pore liquids are, on average, migrating upwards. Fitzmaurice et al. (1995) draft, summarize as follows:

> The near surface data plot to the left of the equilibrium line and indicate an upward liquid migration from an average depth of 100 ft (30 m) to the lands surface. The upward liquid migration suggests long-term evaporation and drying in this region. ... Ranging from 75 to 125 ft (23 to 38 m) below the land surface, a transition zone is present which separates the region of upward flux from the region of downward flux. This region is termed the zero flux plane because water potential plot on the equilibrium line.

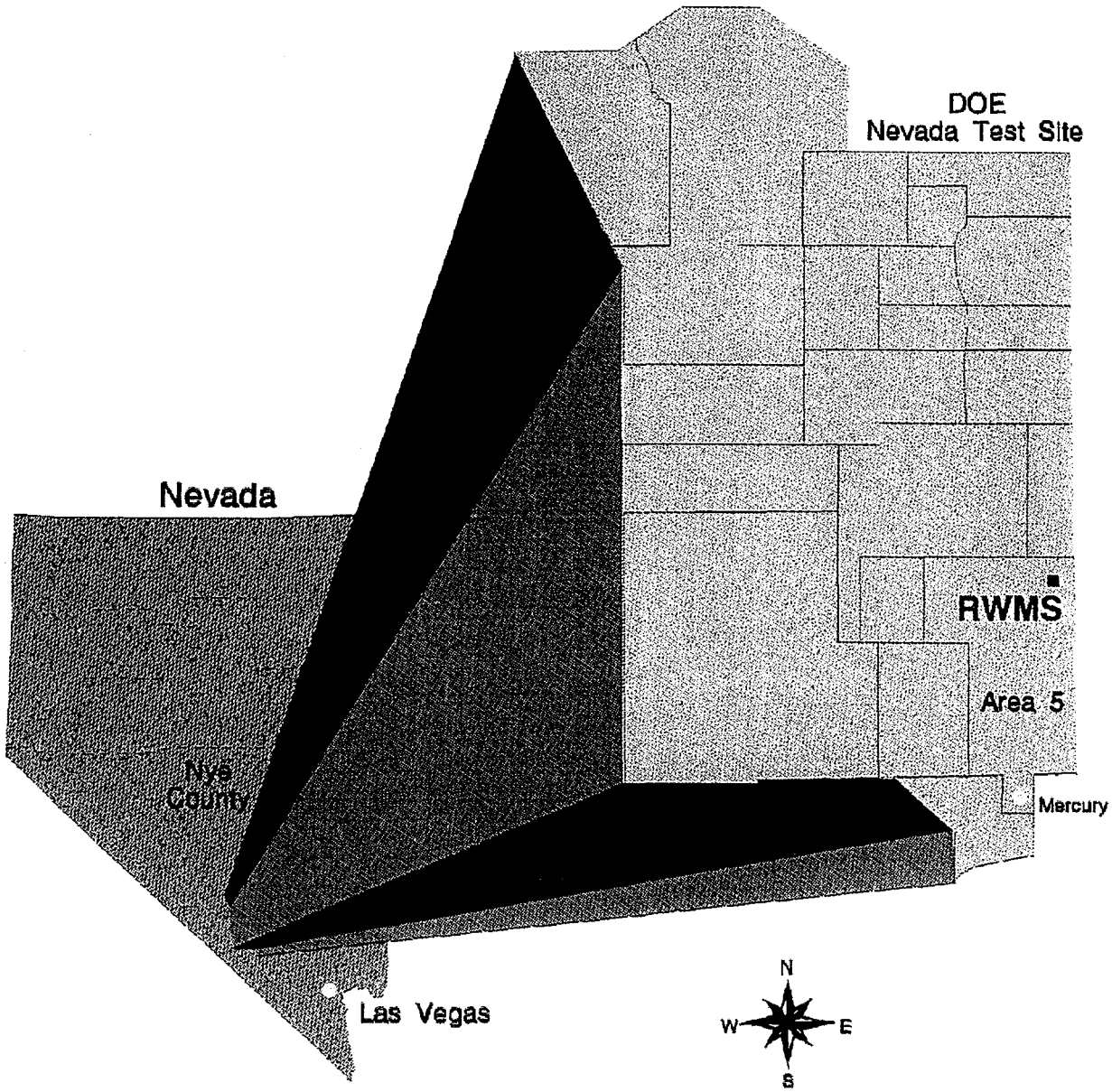

Figure 2. Area 5 RWMS location map.

In summary, the GCD boreholes are located in an arid alluvium, with a deep water table and, on average, pore water that moves upward in the upper 100 ft (30 m) of the sediments.

4.2 Background on Greater Confinement Disposal Concept

The DOE is responsible for disposing of a variety of radioactive wastes, including high-level, TRU and LLW. According to the Nuclear Waste Policy Act of 1982 and the Nuclear Waste Policy Amendments Act of 1987, HLW is potentially destined for the proposed repository at Yucca Mountain in Nevada, TRU waste is to be disposed of in the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico, and LLW is generally disposed of using near-surface burial techniques. However, some of these radioactive wastes do not meet the waste acceptance criteria for any of these disposal facilities. Such wastes are termed special case or orphan wastes, and require an alternative disposal method.

In 1981, the DOE's Defense Low-Level Waste Management Program asked DOE/NV to demonstrate the feasibility of the GCD concept for disposal of LLW and certain high-specific activity LLW (i.e. tritium) in an arid region. The GCD concept was so named because it provides greater confinement than near-surface burial. This greater confinement is needed for wastes for which near-surface burial is not an appropriate disposal method. Located in the RWMS of the NTS, this disposal method consists of boreholes approximately 10 ft (3 m) in diameter and 120 ft (36.6 m) deep. Waste is emplaced in the bottom 50 ft (15.2 m) of the borehole and the remaining 70 ft (21.3 m) from the top of the waste to the land surface is backfilled with native alluvium. The boreholes are not capped or lined, and are within the unsaturated zone. The bottom of each borehole is approximately 650 ft (198 m) above the water table (see Figure 1).

A total of 13 GCD boreholes have been constructed to date. The first borehole was a test and received approximately 1,000,000 curies of high-specific activity LLW. The test GCD borehole was successful in isolating the LLW from the surface environment and an additional 12 GCD boreholes were constructed. Four of those boreholes are unused, and DOE/NV has disposed of LLW, RCRA hazardous, and TRU wastes in the remaining eight boreholes. To assess the likelihood that this TRU waste will comply with the requirements of 40 CFR 191, which is the EPA's standard for the disposal of TRU, high level and spent fuel wastes, Sandia is applying an iterative PA approach based on the high level waste PA methodology developed by Sandia.

4.3 Waste Characteristics

History of the silos and silo residues (Hamric, 1994)

> The silo residues have been stored at the FEMP in Silos 1, 2, and 3 since the early 1950s. The residues were generated by the processing of pitchblende ores and uranium concentrated ores at the FEMP and other DOE facilities for their uranium content. The resultant uranium metal was used in direct support of the United States defense programs. Silos 1 and

2, also known as the K-65 silos, contain "hot" raffinates, also known as K-65 residues. The residues were termed "hot" raffinates because they contain elevated concentrations of gamma emitting progeny from the radium-226 decay series. Silo 3 contains metal oxides, also known as "cold" metal oxides because they contain comparatively lower concentrations of gamma emitting progeny from the same decay series than the K-65 residues. Silos 1, 2, and 3 along with the empty Silo 4 are part of Operable Unit 4.

Silos 1 and 2 contain residues generated from the extraction of uranium from pitchblende ores at the Malinckrodt Chemical Works in St. Louis, Missouri and the FEMP. The majority of these ores originated from the Shinkolobwe Mine in the Belgian Congo with the balance coming from the Rum Jungle Mine and Radium Hill Mine in Australia. As generation of the residues continued at Malinckrodt Chemical Works, storage became a problem. Therefore, the K-65 residues were initially sent to Lake Ontario Ordinance Works near Niagara Falls, New York for storage. Some of these drums were emptied into a tower at Lake Ontario Ordinance Works and the remaining drums were sent to the FEMP. Later, the residues were sent directly from Malinckrodt Chemical Works to the FEMP.

At the FEMP, the drums were transferred into Silos 1, and 2 via the K-65 Drum handling Building. In the Drum Handling Building, the drums were emptied in a tank and mixed with water. The resultant slurry was pumped into Silos 1 and 2, where the solid and liquid separated. The liquid was decanted into the Decant Sump Tank via ports on the silos' sides leaving behind the wet solid which remains in Silos 1 and 2 today. The Decant Sump Tank also contains K-65 sludges which were generated as solids settled or precipitated from the decanted Silos 1 and 2 liquids.

Silo 3 contains residues known as "cold" metal oxides. These residues were generated only at the FEMP. Before being placed into Silo 3 the residues were processed through a spray calciner or a rotating dryer. At the spray calciner, liquids were evaporated at a temperature of approximately 510°C (950°F). At the rotating dryer, some liquids were removed by centrifugal force, and the remaining liquids were evaporated in a rotary calciner operating at approximately 650°C (1200°F) to 820°C (1500°F). During the calcining, the metal nitrates were converted to metal oxides. Finally, the fine metal oxide powder resulting from the calcining was pneumatically transferred to Silo 3.

Characteristics of the untreated silo residues (Hamric, 1994)

The primary chemical contents of the K-65 residues are a mixture of hydroxides, carbonates and sulfates. The radionuclides in the K-65 residues includes uranium (U)-238 and progeny [including U -234, thorium (Th)-230, radium (Ra)-226, lead (Pb)-210, and polonium (Po)-210]; U-235

and progeny [including protactinium (Pa-231) and actinium (Ac)-227]; and Th-232 and progeny (including Th-228). Of these radionuclides, those with the relative highest concentrations are Ra-226 (400000 picoCuries per gram (pCi/g)), Po-210 (194000 pCi/g), and Pb-210 (156000 pCi/g). Due to the high Ra-226 concentrations the K-65 residues emanate radon-222 at a rate of approximately 4500 $pCi/m^2/s$. The predominant inorganic constituents are lead (66200 mg/kg), calcium (17000 mg/kg), iron (15500 mg/kg), and barium (9450 mg/kg). Twenty-five organic constituents were detected with only tributyl phosphate (22 mg/kg), Aroclor-1254 (7.0 mg/kg) and Aroclor-1260 (2.0 mg/kg) exceeding a mean of greater than 1 ppm.

The volume of the K-65 residues in Silo 1 is approximately 4293 yd^3, and Silo 2 contains approximately 3719 yd^3. In 1991, a removal action was performed which added a layer of bentonite clay to Silos 1 and 2. This was done to reduce the radon-222 emissions from the silos. A total of 878 yd^3 of bentonite clay was added bringing the total volume of material for treatment and disposal in Silos 1 and 2 to approximately 8890 yd^3.

The primary chemical contents of the byproduct material in Silo 3 are a mixture of oxides. The radionuclides in the Silo 3 residues include U-238 and progeny (including U-234, Th-230, Ra-226, and Pb-210); U-235 and progeny (including Pa-231 and Ac-227); and Th-232 and progeny (including Ra-228, Th-228, and Ra-224). Of these radionuclides, those with the highest concentrations are Th-230 (51200 pCi/g), Ra-226 (2970 pCi/g), and Pb-210 (2620 pCi/g). The predominant inorganic constituents are magnesium (58600 mg/kg), iron (37800 mg/kg), sodium (36100 mg/kg, calcium (29400 mg/kg), aluminum (17200 mg/kg), and potassium (7260 mg/kg). Although no analytical data is available for the organic constituents in Silo 3, their existence in the oxides is unlikely due to the high temperatures of the calcining performed during generation. The total volume of the residues in Silo 3 is approximately 5088 yd^3.

As part of the Operable Unit 4 Remedial Investigation, samples from the silos were tested via the Toxicity Characteristic Leaching Procedure (TCLP) and Extraction Procedure Toxicity test for Toxicity Characteristic metals. These results indicate that lead (570 mg/L) exceeded the RCRA regulatory limits for the K-65 residues and that arsenic (9.4 mg/L), chromium (5.1 mg/L), and selenium (2.7 mg/L) exceeded the regulatory limits for the metal oxides. Although the untreated silo residues exceed the RCRA regulatory limits for these toxicity characteristic metals, they do not meet the definition of a RCRA hazardous waste. The residues are defined as byproduct material under the Atomic Energy Act and are therefore excluded from the definition of hazardous waste under RCRA. Despite this exclusion, the silo residues will be stabilized through vitrification. This treatment effectively reduces the leachability of the

toxicity characteristic metals below RCRA limits and is described in the following section.

4.4 Waste Treatment, Certification and Disposal (from Hamric, 1994)

The following information is from Hamric (1994). The Record of Decision for the Fernald Operable Unit 4 was issued by the U.S. EPA on December 7, 1994. The OU 4 Record of Decision identifies vitrification as the selected remedy. Treatment (vitrification) of the wastes is necessary for a number of reasons. First, because the risks associated with the untreated wastes exceed generally accepted regulatory thresholds and also because the untreated wastes do not meet the NTS Wastes Acceptance Criteria identified by NVO-325 (the untreated waste fails to meet the leaching criteria and includes free liquids).

The heart of the vitrification system is the melter, which is a refractory-lined cavity with submerged electrodes. After preheating the cavity and the initial charge, the melt becomes electrically conductive. At this time the molten glass is heated by an alternating current generated between pairs of electrodes immersed in the melt. Chemicals may be added to the residues, as required, to aid in achieving suitable durability or processability. The additives may include reagent grades of silicon oxide (SiO_2), alumina (Al_2O_3), boric acid (H_3BO_3), sodium carbonate (Na_2CO_3), and carbon. The molten glass may either be cast into monolithic shapes, formed into smaller shapes, or quenched to a frit.

Vitrification is not an innovative technology; however, the application of the vitrification technology to the treatment of hazardous and radioactive waste is innovative. Treatability studies and benchscale tests performed by OU4 have shown that the silo residues are readily amenable to the vitrification technology. The K-65 residues, in particular, easily form a good glass without significant additives or process adjustments. The selection of vitrification as the preferred remedial treatment technology for OU4 materials was based on a significant quantity of process knowledge and data accumulated from the extensive OU4 vitrification treatability studies.

Characteristics of the vitrified product

Materials to undergo vitrification include the residues stored in Silos 1, 2, 3 and sludges in the Decant Sump Tank (estimated volume of sludge is 5 yd^3). If any soils are encountered during remediation of OU4 which exhibit highly elevated direct radiation levels as a result of contamination from the silo residues, those soils will also be vitrified. The total volume of the residues prior to treatment will be approximately 14,000 yd^3. Information from laboratory scale treatability tests indicate that the vitrification process will reduce the volume of material for disposal to approximately 6000 yd^3.

It is expected that the total volume of soils to be vitrified would not be significant in comparison with the total volume of the silo residues.

This treatment effectively reduces the leachability of toxicity characteristic metals below RCRA regulatory limits, and creates an overall more stable residue form which is more compatible for disposal. In particular it reduces radon emanations from the K-65 residues from approximately 4500 $pCi/m^2/s$ to less than 1 $pCi/m^2/s$.

Evaluations performed by OU4 personnel in conjunction with the FEMP Waste Characterization group indicate that the vitrified residues would meet the "General Waste Form Criteria" as listed in Section 5.5.1.1 of NVO-325. However, after vitrification, the glass will be tested as confirmation that concentrations of RCRA Metals no longer exceed the RCRA toxicity characteristic limits and that radionuclides are properly immobilized within the glass. Measurements will also be performed to indicate that the radon emanation rates are effectively attenuated. Goals for these levels will be driven by NTS waste acceptance criteria. Results of the analysis will be evaluated and forwarded to NTS, as required, prior to shipment of the wastes to NTS for disposal. In addition, all sampling and analysis will be performed per NVO-325 requirements, including quality assurance (QA) and use of standardized data reporting forms.

Remediation schedule

The scheduled date for beginning the removal and treatment of the residues is March 1997. Therefore, assuming a three month testing and evaluation period, shipments would begin in June 1997 and would continue on a regular basis until June 2000.

4.5 Assumptions About Waste Form

According to Bob Vogel of FERMCO (telephone conversation of February 1, 1995 with H.W. Stockman, Sandia), the preferred form for the glass waste is flattened "gems," about 4 g in mass and 3/8 in. thick. Current plans call for pouring the gems into 3 gallon cans and placing approximately 168 cans in a 6 in.-thick concrete "square-pack." According to Bill Tope FERMCO (phone conversation February 8, 1995 with H.W. Stockman, Sandia), the external dimensions of the square pack are 85 in. by 62 in. by 69 in.

For the following calculations, the glass is assumed to have a density of $\rho_g = 2.87 g/cm^3$, consistent with the values given by DOE (1993) and FERMCO (1995). Accordingly, a 4 g gem corresponds to a volume of 1.394 cm^3. Assuming the closest-packing geometry gives 25.9% void space for spheres (Moore, 1972).

Approximately 3253 concrete boxes will be required to ship the 6,000 yd^3 (4,580 m^3) of vitrified wastes. Boxes cannot be placed side by side in the 10-ft (3-m) diameter

boreholes, but must be stacked on top of each other, utilizing about 52% of the cross-sectional area of a borehole. Using the existing GCD configuration, 10 boxes can be placed in each borehole. Consequently, a total of 326 GCD boreholes will be required to dispose of the 6,000 yd^3 of vitrified silo wastes.

There will be about 18 yd^3 of solid waste (25 yd^3 of vitrified gems plus voids) inside the 145 yd^3 of space allotted for waste in each borehole.[6,7]

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[6]The draft report stated 3927 yd^3 of space for waste per borehole, which was incorrect. The correct volume is 3927 ft^3 or 145 yd^3, and the correct number was used to estimate the number of GCD boreholes required to contain the waste.

[7]As noted on the Executive Summary, Fernald is reevaluating this treatment option and the choice of waste containers.

5.0 Analysis of Performance

A PA is a compliance calculation, an evaluation of whether or not the potential dose will be below a given performance objective, rather than a calculation, or prediction of actual dose. In this performance evaluation, the doses to the hypothetical member of public and inadvertent intruder are simulated using gas phase and liquid phase transport models to calculate the concentration of radionuclides at the accessible environment as a function of time. The concentrations from the transport models are used as input for the dose models. The wastes from Silos 1 and 2 (K-65 wastes) are evaluated separately from the Silo 3 wastes because of different source concentrations. The following sections contain a detailed discussion of how the source terms for the transport models were estimated, the exposure pathways that were evaluated, the conceptual and numerical models of the transport system, the dose calculation methods and the input parameter values for each of the models.

5.1 Source Terms

The decay chains for the K-65 and Silo 3 vitrified wastes are shown in Figure 3 (top). The half-lives of the parent isotopes (^{238}U, ^{235}U and ^{232}Th) are on the order of 700 million (^{235}U) to 10 billion years (^{232}Th). Because of these long half-lives, it is likely that transport of radionuclides will have to be simulated for very long periods of time in order to estimate the peak concentrations of the isotopes at the accessible environment. For the preliminary evaluation, the isotopes with half-lives less than one year are not modeled. The isotopes that are included in the liquid phase transport model are shown in Figure 3 (bottom). Radon is transported in the gas phase and is modeled separately.

5.1.1 Calculation of Radon Concentration

The current PA radon diffusion models assume a concentration boundary condition at the source region, so it is necessary to estimate concentration in the interstices of the glass marbles.

Assumptions About Waste Form

The glass is assumed to have a density of $\rho_g = 2.87 g/cm^3$. This value is at the high end of the reported range (2.6 to 2.87 in FERMCO, 1995) and was chosen to maximize the estimation of radon emanation rate. A 4 g gem corresponds to a volume of 1.394 cm^3 and a surface area of 6.035 cm^2 (spherical geometry) to 7.536 cm^2 (cubical geometry). The higher density and larger surface area is assumed to maximize radon release rate. The gems are assumed to be closest-packed (25.9% void space, or 0.1218 cm^3 air/g of

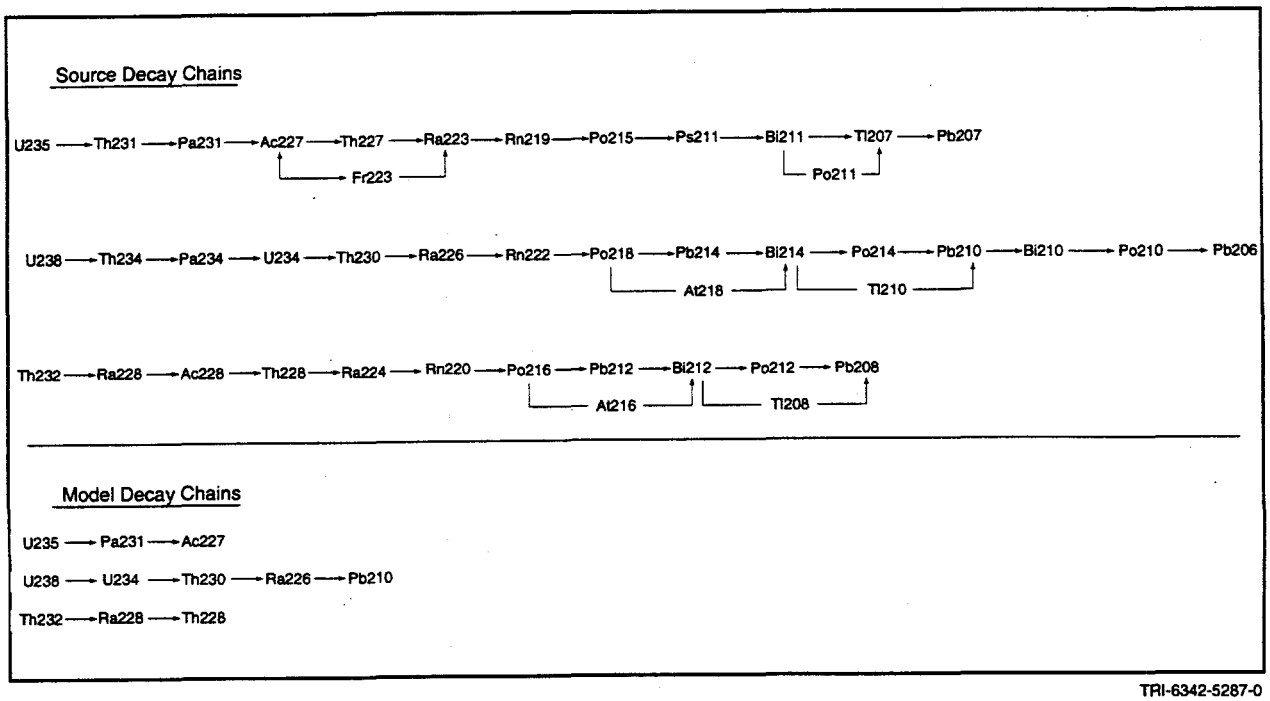


Figure 3. Waste decay chains (top) and decay chains used in the liquid diffusion model (bottom).

glass.), to minimize the volume of air per g of glass, and thus maximize the concentration of emanated Rn in the air.[8]

Decay/Growth Model

The Rn emanation rate is assumed to be proportional to the ^{226}Ra concentration of the glass. The ^{226}Ra concentration of the glass is controlled, for the time period of the calculation, principally by decay of ^{230}Th to produce ^{226}Ra, and the decay of ^{226}Ra to produce ^{222}Rn.. Let:

M = atoms ^{226}Ra per unit mass of glass; M_0 = M(t=0=present time);

[8]A void space of 25.9% is the smallest void space for perfectly packed spheres, and 0% is the smallest void space for perfectly packed cubes; 25.9% was chosen as a reasonably conservative void space for the glass gems, while still maximizing density and surface area.

P = atoms ^{230}Th per unit mass of glass; P_0 = P(t=0=present time);

λ_M = decay constant of ^{226}Ra = $\ln_e 2/(1600 \text{ years} = 5.0492 \cdot 10^{10} \text{ s})$;

λ_P = decay constant of ^{230}Th = $\ln_e 2/(80000 \text{ years} = 2.5246 \cdot 10^{12} \text{ s})$;

m = the activity of ^{226}Ra per unit of glass ($m \propto \lambda_M M$);

p = the activity of ^{230}Th per unit of glass ($m \propto \lambda_P P$).

Table 3, which is modified from Table 2-2 in the report by Foster Wheeler Environmental Corporation (1995), gives the estimated values of the decay model parameters.

Table 3. Radionuclide Parameters for Two Glass Compositions

	K-65 sequence A0.1 glass	Silo 3 sequence C0.1 glass
m_0 in pCi/cm^3 ^{226}Ra	1,328,545	13,059
M_0 in atoms/g ^{226}Ra	$3.581 \cdot 10^{15}$	$3.520 \cdot 10^{13}$
m_0 in pCi/g glass ^{226}Ra	462908	4550
p_0 in pCi/cm^3 ^{230}Th	212,233	394,547
P_0 in atoms/g ^{230}Th	$2.860 \cdot 10^{16}$	$5.317 \cdot 10^{16}$
p_0 in pCi/g glass ^{230}Th	73949	137473

The production and loss of ^{226}Ra follows:

$$dM/dt = \lambda_P P - \lambda_M M; \quad P = P_0 \exp(-\lambda_P t).$$

It is easily shown that the solution to these equations is:

$$M(t) = [\lambda_P P_0 / (\lambda_M - \lambda_P)][\exp(-\lambda_P t) - \exp(-\lambda_M t)] + M_0 \exp(-\lambda_M t).$$

The atoms of ^{222}Rn in the air in the interstices of the gems will follow:

$$d(Rn)/dt = k - \lambda_{Rn} \cdot Rn$$

where k is the emanation rate, and λ_{Rn} is the decay constant for ^{222}Rn. After about 6 half-lives of ^{222}Rn (23 days),

$$d(Rn)/dt \approx 0 \Rightarrow Rn \approx k/\lambda_{Rn}, \text{ or the (activity of Rn in air)} = \lambda_{Rn} Rn = k$$

for the control volume.

It is next necessary to relate the activity of ^{222}Rn in the interstices to m(t). We assume:

$$k(t) = k_0\, m(t)/m_0.$$

It is most convenient to consider the emanation rate (pCi of ^{222}Rn) per g of glass, and consider the control volume of air to be the volume (0.1218 cm^3) associated with 1 g glass in the closest-packing geometry. The highest emanation rate given on page 54 in DOE (1993) is 0.059 pCi/(m^2 s) at 30 days after fabrication (sample A0.1). From table 2.2 in Foster Wheeler Environmental Co. (1995), this material contains 1,328,545 pCi/cm^3 of ^{226}Ra at present. Converting to consistent units, we obtain a concentration of radon per cm^3 of air:

$$C_{Rn}(t) = \frac{K_0}{\lambda_{Rn}} \cdot \frac{m(t)}{m_0} \cdot \text{unit conversion factors}$$

$$= \frac{0.059 \ \frac{\text{pCi Rn}}{\text{m}^2\cdot\text{s}}}{2.097 \times 10^{-6}/\text{s}} \cdot \frac{1}{1328545 \ \frac{\text{pCi Ra}}{\text{cm}^3}} \cdot \frac{7.536 \times 10^{-4}\,\text{m}^2}{4 \text{ g glass}} \cdot \frac{1 \text{ g glass}}{.1218 \text{ cm}^3}$$

$$\cdot m(t) \cdot \frac{2.87 \text{ g}}{\text{cm}^3}$$

$$= \frac{9.39 \times 10^{-5} \ \frac{\text{pCi Rn}}{\text{cm}^3}}{\frac{\text{pCi Ra}}{\text{g glass}}}$$

$$C_{Rn}(t) = 9.388\cdot10^{-5}\ [(\text{pCi Rn})/(\text{cm}^3 \text{ air})/(\text{pCi/g glass})] \cdot m(t).$$

Figure 4 shows the calculated radon concentration in the air in the interstices of the two glasses as a function of time.

5.1.2 Calculation of Ra, Pb, U, and Th Solubility

The bulk solubilities of radium, lead, uranium, and thorium were calculated with version 7.2a of the EQ3/6 code suite (Wolery and Daveler, 1992) and the PP postprocessor program (Stockman, 1994). Two idealized glass compositions (Table 4), corresponding to the sequence A and C test compositions of DOE (1993), were reacted with two pore water compositions in both titration and flow-through models (nmodl1 = 1 and 3 in EQ6 language). The two glass compositions correspond approximately to the glass types in

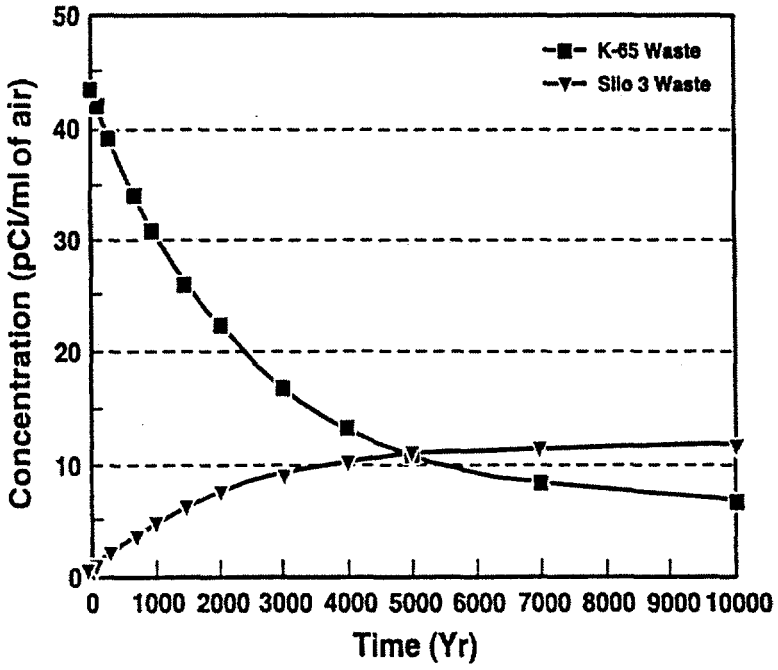


Figure 4. Radon source concentration for K-65 and Silo 3 wastes.

Table 3, and also represent the end members of the composition range investigated by Foster Wheeler Environmental Corporation (1995). The compositions were derived from Table 4.18 in DOE (1993), but were modified in five ways: (1) Ni, Co, and Cu were added into the Fe content; (2) 1% sulfur was added to each composition (sulfur was not analyzed for Table 4.18 of DOE (1993), but was clearly present in abundance); (3) an amount of Ra, corresponding to radiochemical analyses of the samples reported in DOE (1993), was added to the compositions; (4) Th was added to the sequence A composition; even though Th is not reported in Table 4.18 of DOE (1993), leaching analyses show it is clearly present; and (5) small amounts of rare earths and a few other metals were eliminated, due to lack of thermochemical data and lack of radiological importance.

The two pore water compositions are referred to as "cement-equilibrated" and "extreme GCD-area alluvial pore water." The former is intended to reflect the possible influence of the "square pack" concrete containers, and was created by using EQ6 to titrate Portland cement (composition taken from Lea, 1970) into water until saturation was achieved, at a CO_2 pressure of $10^{-3.5}$ atmospheres. Fixing the CO_2 pressure forces the cement-equilibrated solution to a modest pH; however, we wish to model the solution derived by leaching cement with water in an arid environment, not the solution that exists in the pore waters of concrete. According to Walton et al. (1990), the carbonation rate of cements in the vadose zone of arid environments is several orders of magnitude faster than the leaching rate of $Ca(OH)_2$, so we expect modest pH and

Table 4. Idealized glass compositions

Component	Weight % in Sequence A Glass	Weight % in Sequence C Glass
SiO_2	54.3	30
Na_2O	15.2	6
PbO	10.6	0.19
FeO	4.9	9.4
MgO	1.5	10.2
Al_2O_3	3.2	20
BaO	5.4	0.03
P_2O_5	0.73	9.4
CaO	1.3	4.7
K_2O	0.75	1.8
UO_2	0.19	0.34
ThO_2	0.19	0.32
B_2O_3	0	5
RaO	0.0000316	0.000000113
S	1	1

equilibration with CO_2 in waters leached from the square pack. The "extreme GCD-area alluvial pore water" composition was derived from leaching analyses reported in Stockman et al. (1993), and is 0.024 molar in sulfate, 0.14 molar in Cl^-, and 0.00021 molar in F^-. The latter ions can influence both the complexation and precipitation of radionuclides, so the solution composition was chosen to reflect the extreme values found from leaching tests.

Initial reaction path modeling showed several problems with the EQ3/6 data0.com.R22a database. We used a modified version of the database (referred to as data0.com.HWS1 in our log books); this database includes notes and references for the changes made by Sandia. In brief, the problems we encountered included: (1) incorrect equilibrium constants for several polynuclear aqueous silica species; (2) inaccurate data for solid $BaCO_3$; (3) a lack of data for solid magnesium and thorium phosphates; and (4) a lack of data for solid $RaCO_3$. In addition, it was necessary to add solid solutions models for the $(Ba,Sr,Ra)SO_4$ and $(Ba,Sr,Ra)CO_3$ series.

The results of the final reaction path modeling are shown in Figures 5 through 8. All runs assumed an O_2 pressure of 0.2 atmospheres and a CO_2 pressure of $10^{-3.5}$

atmospheres, the ambient for normal air. Because of time constraints, we used relative glass dissolution rates for our reaction path models (a more realistic transition-state dissolution model is under development at Sandia, but will not be available until April, 1995). The full x-axis range in the top plot of Figures 5 through 8 represents 1 g of glass altered per g of pore water. Assuming a pH of 8 to 10, and the rates given in Knauss et al. (1990), this amount of alteration would represent reaction over approximately 500 years. However, the rates from Knauss et al. (1990) reflect dissolution into very dilute water, and could greatly overestimate the true dissolution rate, since they reflect neither the formation of coatings on the glass surface nor the saturation of the solution with glass components.

In our reaction path modeling, glass was added until the solubilities of Ra, U, Th, and Pb ceased to climb and the system was near saturation with all but the most soluble components. There is a slight positive slope for the solubility in some plots, but this effect is due to the increasing ionic strength (near 1 molal in some plots at the end of the run), and the subsequent suppression of the activity coefficients for complexes. It is unlikely that the interstitial solutions could ever achieve an ionic strength of 1 molal due to diffusion and influx of pore water from the surrounding alluvium. The peaks and jags in the plots reflect the underlying mineral transitions, which were quite complicated; between 16 and 20 distinct minerals and solid solutions formed in each run. The high peak in Ra concentrations in Figure 5 (top) is due to the transient formation of $(Ba,Ra)CO_3$ solid solution, in preference to $(Ba,Ra)SO_4$ solid solution.

Figures 5 and 6 show results for two distinct reaction path models; a titration model (filled symbols), where the precipitated alteration minerals are allowed to continuously react with the pore solution, and an open-system model (open symbols) where the alteration minerals are prevented from further reaction with the solution once formed. The latter model may be the more realistic. In any case, the two models agree well near the end of each reaction path run. The only significant discrepancy between the two models is shown in Figure 5, where the open-system model yields a much smaller increase in Ra concentration from the formation of $(Ba,Ra)CO_3$. The thorium solubility is higher in the sequence C glass (Figures 6 and 8), because it has higher phosphate concentrations than the sequence A glass. When the sequence C glass dissolves, there is not enough calcium, magnesium, and lead to tie up all of the phosphate. Consequently, the phosphate complexes with thorium as an aqueous species.

Due to the time limitations, a deterministic analysis is used in this evaluation, and a single bulk solubility value is required for each of the elements. The source terms for this evaluation are based on the arbitrary assumption that 0.1 g of glass will dissolve in 1 g of water at the source. The solubility of the individual isotopes (S_i) in the liquid phase transport model is a function of the bulk solubility for the element (S_e) and the relative abundance of the isotope.

$$S_i = S_e \text{ (moles of i) / (total moles of isotopes of e)}$$

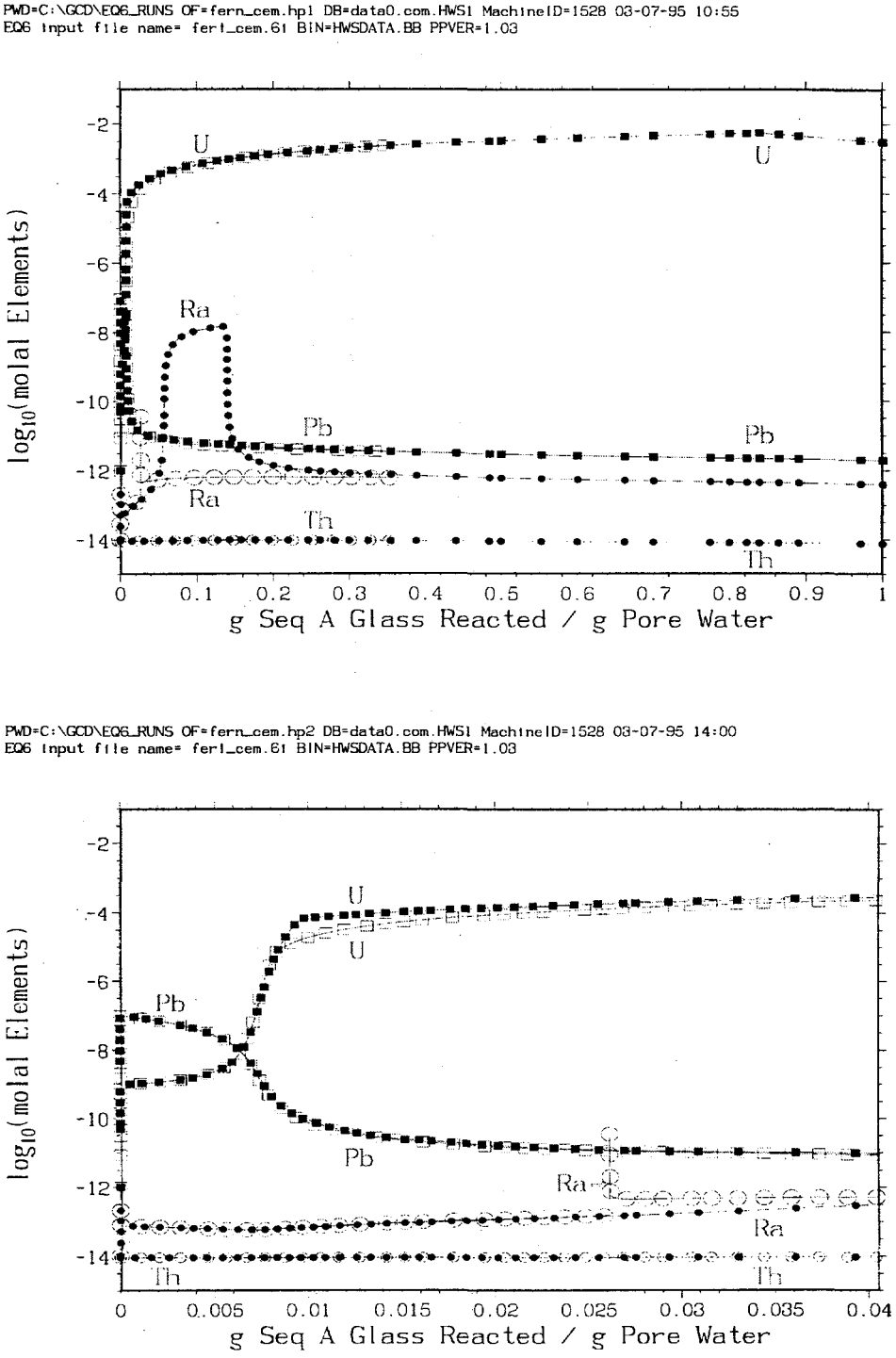


Figure 5. Solubilities for sequence A glass in cement-equilibrated water. Open symbols are for open-system (flow-through) model; filled symbols are for titration model. Bottom figure is an expansion of top for small x.

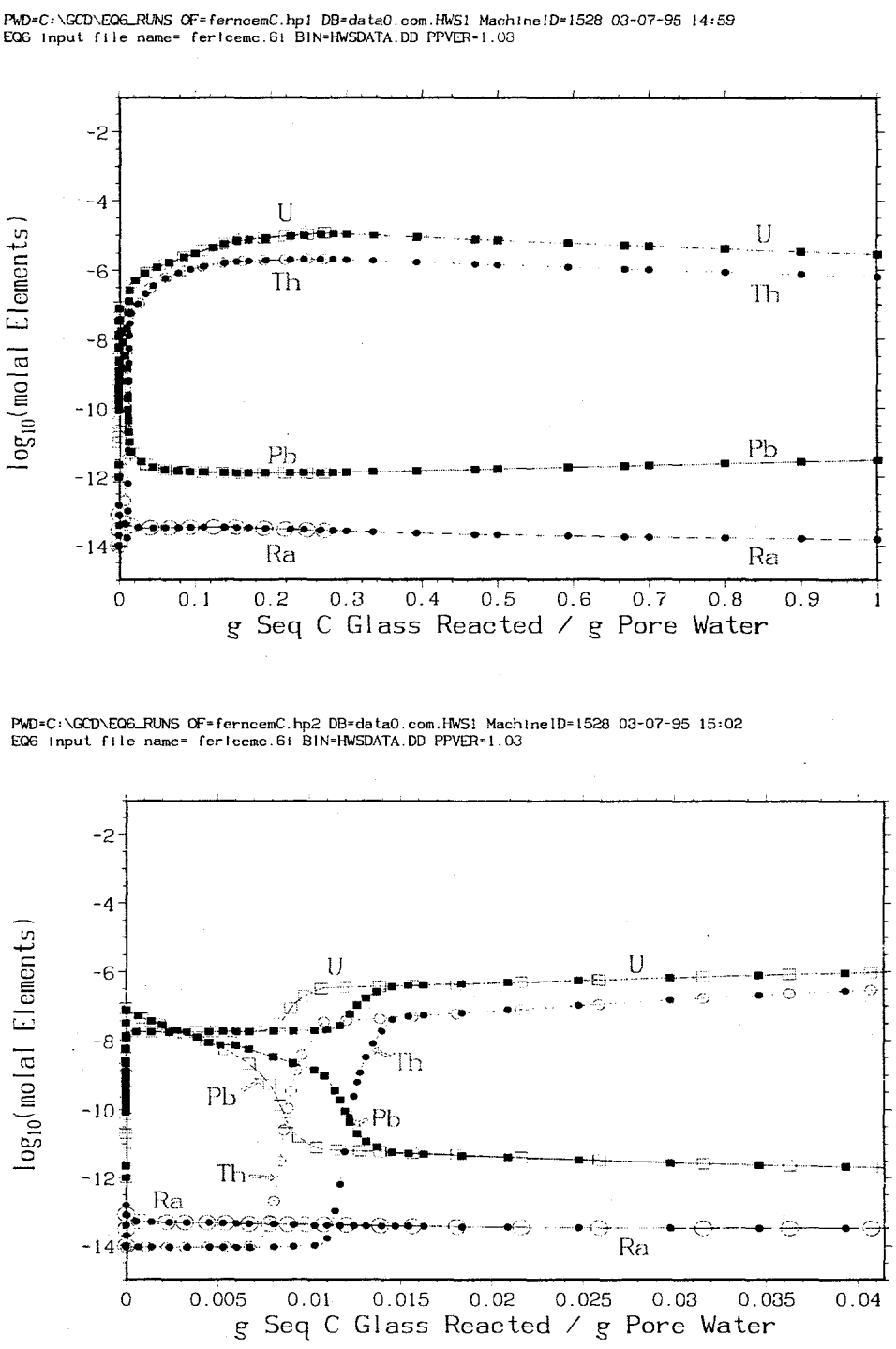


Figure 6. Solubilities for sequence C glass in cement-equilibrated water. Open symbols are for open-system (flow-through) model, filled symbols are for titration model. Bottom figure is expansion of top for small x.

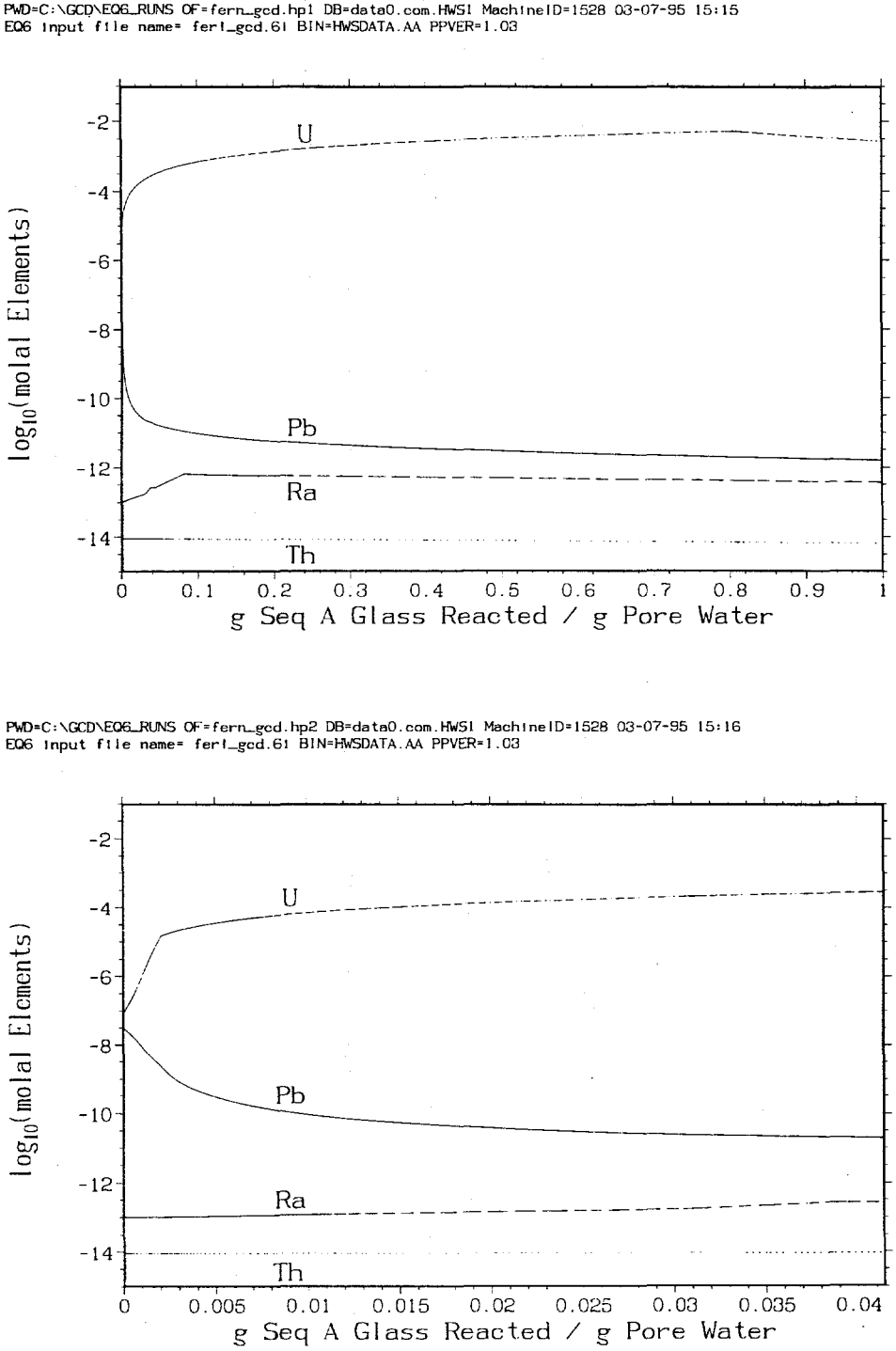
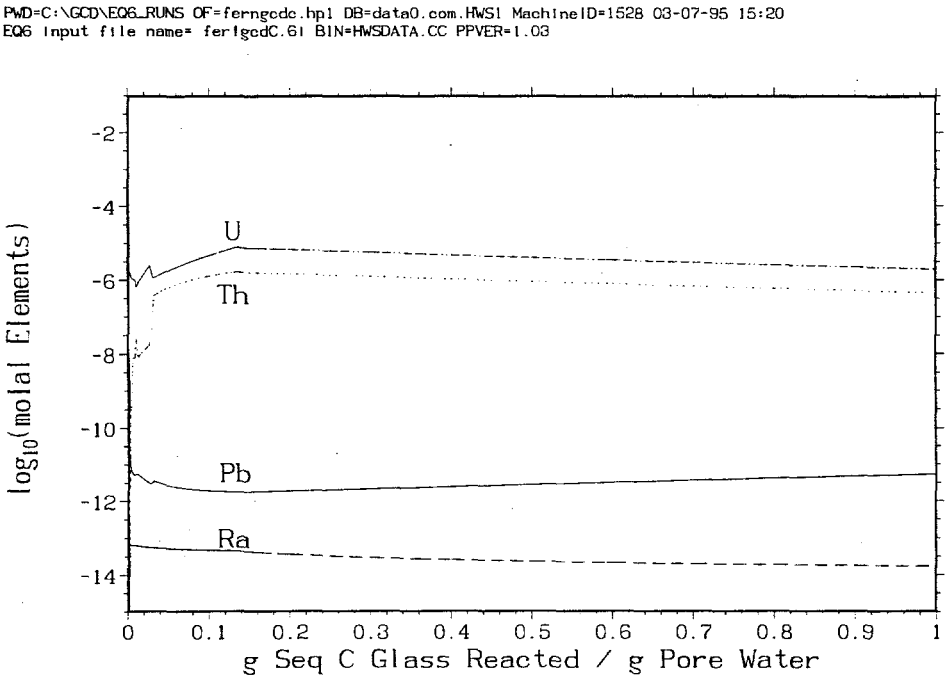


Figure 7. Solubilities for sequence A glass in "extreme" GCD-area alluvial pore water. Bottom figure is an expansion of the top for small x.



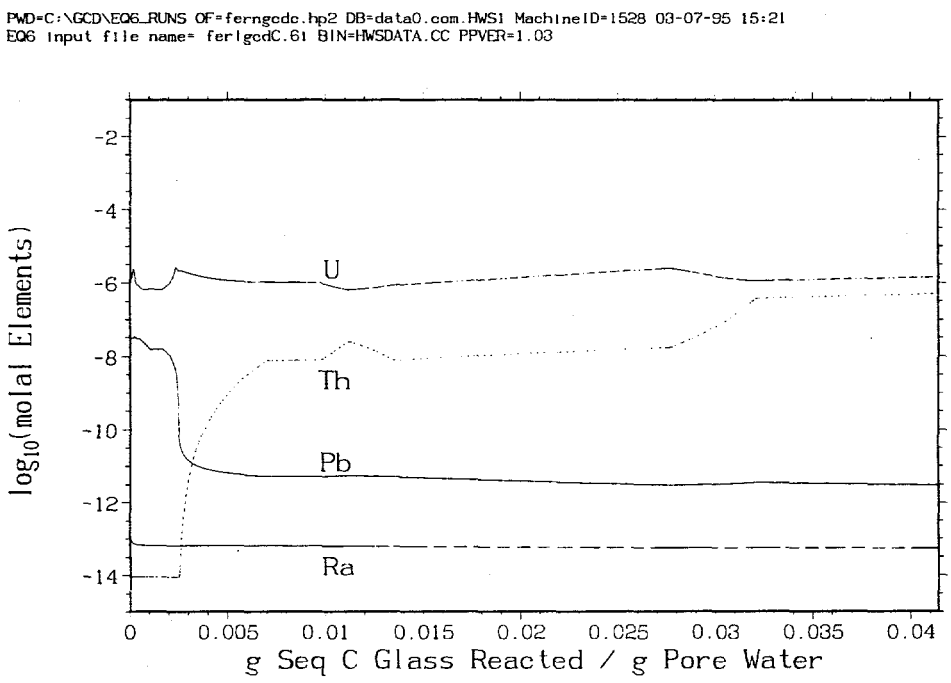


Figure 8. Solubilities in sequence C glass for "extreme" GCD-area alluvial pore water. Bottom figure is expansion of top for small **x**.

The isotope solubilities for the liquid diffusion model were calculated using a spread sheet program written for this evaluation. The output from the program is shown in Appendix 2.

The solubilities of two of the elements included in the modeled decay chains, actinium and protactinium, were not simulated because of a lack of thermodynamic data on these elements. Consequently, the bulk solubilities of actinium and protactinium that were used in the second PA for the GCD site (Baer et al., 1994) were used in this evaluation. These solubilities are believed to be conservatively high.

The source term for the liquid phase transport model assumes solubility-limited concentration. The source concentration is a function of the solubility and amount of the isotope in the borehole at the source. The amount of waste in each borehole is estimated based on the assumption that the K-65 and Silo 3 wastes are separate (except in one borehole) and that the waste from each source is of uniform composition. There is one borehole that will contain only three concrete boxes and one of those boxes is assumed to contain K-65 and Silo 3 wastes (to minimize the number of boreholes). Due to time limitations, this borehole is not included in the performance evaluation. The amount of the individual isotopes, in the full boreholes containing only K-65 or Silo 3 waste, is calculated using a spreadsheet program written for this evaluation. The output from this spreadsheet program is shown in Appendix 2.

5.2 Pathways and Scenarios

This section addresses the time periods of concern, and the exposure pathways for the MOP and inadvertent intruder scenarios.

5.2.1 Time Periods

As discussed in Section 5.1, the half-lives of the parent radionuclides are on the order of millions to billions of years. Figures 9 and 10 show how the activity of the parent radionuclides in the boreholes will decline over these very long time scales. It is reasonable to assume that the peak dose to the intruder and MOP will occur during the period of highest concentration at the source (i.e. sometime within the first 100 million years). The actual time or period during which the peak dose is likely to occur will be a function of the transport processes at the site. It is important to recall that the physical system is certain to change over these long time scales due to natural changes in the climatic conditions.

5.2.2 Pathways

As described in Sections 3.1 and 3.2 there are two scenarios that are evaluated in this performance analysis. This performance evaluation estimates the dose to a hypothetical MOP, who lives at the site but does not disturb the waste and only minimally disturbs the sediments above the waste (upper 6 in. of soil); and the dose to

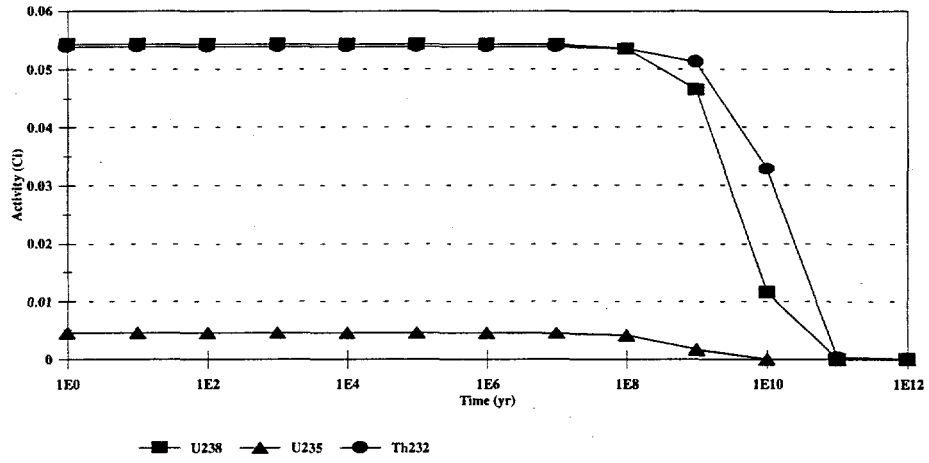


Figure 9. Activity of parent isotopes: K-65 borehole.

a hypothetical inadvertent intruder who lives at the site, disturbs the sediments above the waste by excavating a basement and growing a garden and brings some of the waste to the surface by drilling a well through one of the boreholes.

The potential pathways of exposure to the MOP are limited to inhalation of contaminated surface sediments, direct (external) exposure to radiation from contaminated surface sediments, and internal exposure from ingesting contaminated groundwater. The MOP may also receive a dose from inhalation of radon gas, but this pathway is explicitly excluded in the regulations for the MOP; therefore, it is not included in the dose calculation for the MOP.

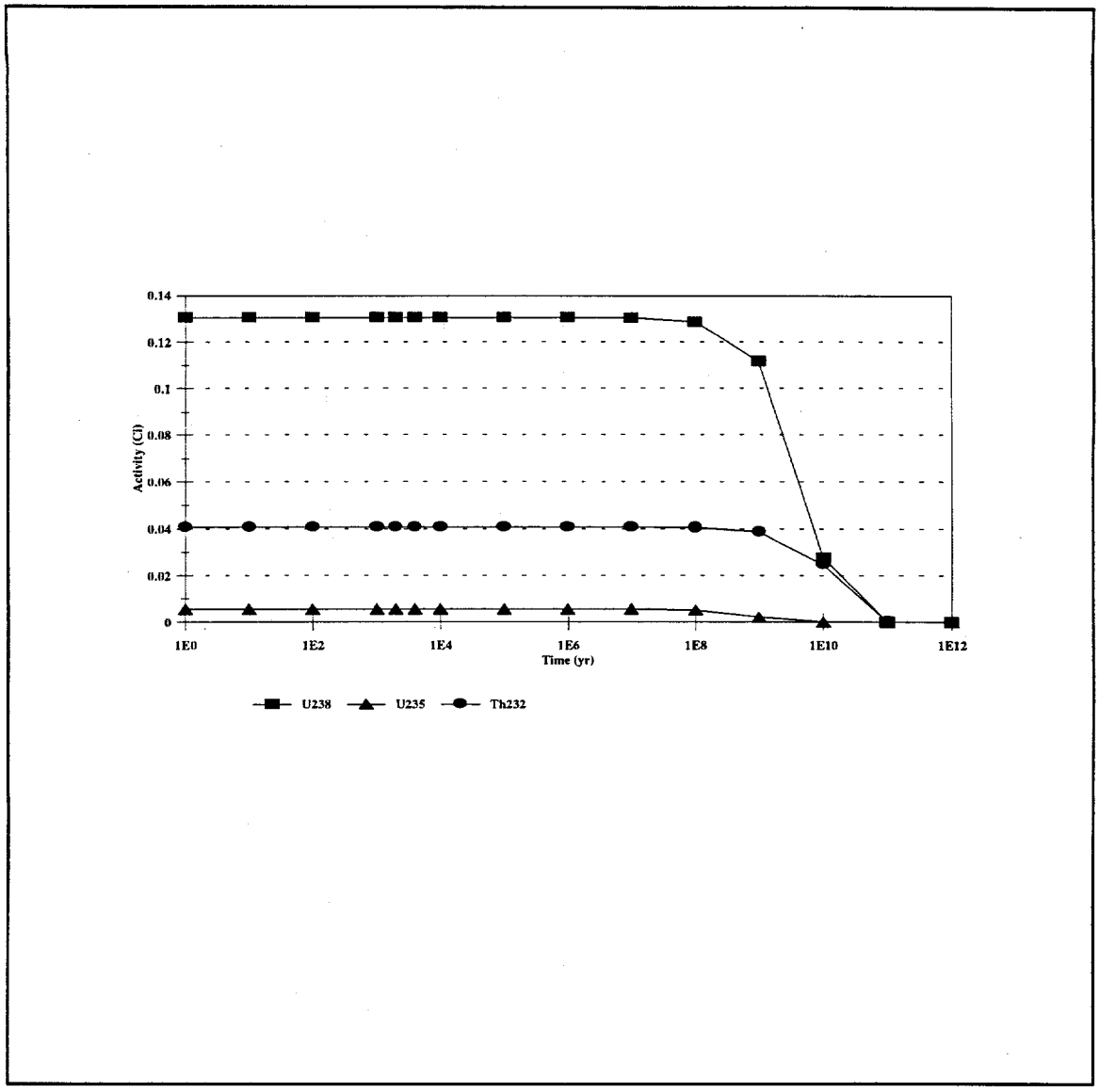


Figure 10. Activity of parent isotopes: Silo 3 borehole.

The intruder has the same potential exposure pathways as the MOP, plus ingestion of contaminated plants and soil and inhalation of waste particles from the well drilling and the tailings pile from basement construction. The dose limit for the intruder includes the dose due to inhalation of radon gas. Consequently, the performance evaluation includes the inhalation of radon (while the intruder is inside and outside the house at the site) in the estimated dose for the intruder.[9]

5.3 Performance Analysis Methodology

[9]Inclusion of radon in assessing doses to the intruder is controversial; to be conservative, we have included radon in the dose assessment for the intruder.

Most of the exposure pathways listed in the previous section are a function of the contamination of surface and near surface sediments with radioactive isotopes. The MOP and intruder receive a dose due to inhalation, ingestion, or direct contact with contaminated sediments. The intruder may also ingest plants that have removed the contaminants from the sediments. In order for these pathways to be significant, the contaminants must move upward from the source toward the ground surface by molecular diffusion, upward advection, or human activity (i.e., well drilling, irrigation with contaminated groundwater, tilling, basement excavation).

Previous PA analyses of the GCD site indicate that, without human intrusion, upward diffusion is the most likely process by which contaminants will reach the accessible environment under existing hydraulic conditions (Baer et al., 1994). Based on the work of Baer et al. (1994), it was assumed that surface and near surface pathways are the most likely pathways of exposure for the MOP and intruder. Although recharge is not significant for the climate conditions that have existed at the site for the last 15,000 to 260,000 years,[10] these conditions are not be expected to continue over the time periods of concern for this waste. However, because there is tremendous uncertainty in what the long term climatic and hydraulic conditions will be and because a deterministic analysis is being used (due to the time and budget limitations imposed on this evaluation), the groundwater pathway and the effects of upward advection are not included in this evaluation.

As a result of these assumptions and simplifications of the system, the performance analysis consists of six steps for each of the waste types (K-65 and Silo 3):

1. The gas phase diffusion of ^{222}Rn is modeled and the flux and air concentration in the sediments at the ground surface and basement elevation are simulated as a function of time,

2. The liquid phase diffusion of ^{238}U, ^{231}Pa, ^{227}Ac, ^{235}U, ^{234}U, ^{230}Th, ^{226}Ra, ^{210}Pb, ^{232}Th, ^{228}Ra and ^{228}Th are simulated and the concentration of each isotope in the sediments at the ground surface, basement depth, tilling depth, and maximum rooting depth are estimated as a function of time,

3. The concentrations from the liquid phase diffusion model are used as input to the GENII code to model the MOP and intruder's exposure to the contaminants,

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[10]Recharge studies have been conducted at the site to estimate the existing recharge rate and the potential increase in recharge due to climate change (Tyler et al., 1996; Tyler et al., 1995; Conrad et al., 1993; Conrad, 1993; Detty et al., 1993; Strong and Conrad, 1992). The results of these studies indicate that under the current climatic conditions there is a net loss in moisture from the subsurface sediments. The distribution of chloride in the unsaturated zone indicates that chloride is assimilating in the upper 125 m (410 ft) of the unsaturated zone and that the system is still responding to the change from wetter conditions, with net infiltration, to conditions where evapotranspiration exceeds precipitation resulting in net discharge from the unsaturated zone. The fallout of ^{36}Cl from weapons testing is located in the upper 2 m (6.6 ft) of the unsaturated zone and the stable isotope composition of the soil water, near the ground surface, indicates evaporative enrichment. The stable isotopes indicate there is net upward movement of the liquid phase in the upper 8 to 39 m (26 to 128 ft) of unsaturated zone.

4. The dose to the intruder from ^{222}Rn is estimated using a separate, empirical dose model,

5. The doses from all the pathways are summed to yield a total dose for the MOP and a total acute and total chronic dose for the intruder, then

6. The total doses are compared to the performance objectives.

5.3.1 Conceptual model of transport

The conceptual model of contaminant transport at this site is based on the interpretation of site characterization data collected for the GCD PA (see Baer et al., 1994 and Price et al., 1993 for specific references and data) and general concepts of unsaturated flow and transport.

Under the current climatic conditions, the wastes would be located at the bottom of the zone with upward hydraulic gradients, and above the zone with downward hydraulic gradients. There may be infrequent, transient infiltration events that reach the depth of the wastes in the boreholes. These transient events could bring relatively fresh water to the waste packages. Based on ^{36}Cl profile, it is unlikely that even a relatively large infiltration event would cause significant downward advection of the contaminants at the site (Conrad, 1993). Hence, the primary transport mechanisms for moving the contaminants from the source to the accessible environment are molecular diffusion, upward advection, and human intrusion (e.g. drilling).

In order for the contaminants to reach the surface and near surface by liquid or gas phase diffusion, they must be released from the metal canisters, and there must be a continuous path from the source to the surface or near surface (i.e. the accessible environment). In the unsaturated zone, the path for gas phase diffusion is intuitively continuous, although slightly tortuous due to the presence of the sediments and some water-filled pores. The liquid diffusion path exists along the surface of the sediments where a relatively thin film of water coats the grains. Due to the low water contents of the sediments, this is a highly tortuous path that effectively reduces the rate of liquid phase diffusion.

The effects of transient infiltration events on the diffusive transport are uncertain. Increased infiltration would temporarily increase the water content of the sediments and decrease the tortuosity of the liquid diffusion path. It is uncertain to what degree the tortuosity would change and what effect the transient, downward advection of the contaminants would have on the distribution of the contaminants in the subsurface. In this evaluation, it is assumed that upward and downward advection are balanced, resulting in no net change in the distribution of the contaminants due to these

processes.[11] This is a simplifying assumption, and it may not be conservative. This leaves diffusion and human intrusion as the only transport mechanisms.

As the contaminants diffuse from the source into the sediments they may be adsorbed on the sediment surfaces (liquid phase), they will undergo radioactive decay and they may precipitate out of solution as the water chemistry changes. Once contaminants reach the root zone some fraction of them may be removed by plants and transported through plants to the ground surface. The remaining contaminants continue to diffuse toward the ground surface. The gas phase contaminants (radon isotopes) continue to diffuse into the atmosphere, while the liquid phase contaminants accumulate at the surface until they are removed by erosion, human activity or by transient infiltration events. For this evaluation, the model is simplified by assuming that once the contaminants are dissolved, they do not precipitate out of solution. It is also assumed that erosion is negligible.

5.3.2 Mathematical models

Gas phase diffusion

An analytical solution to the diffusion equation is used to estimate the concentration of ^{222}Rn at the ground surface and basement elevation. The governing equation for this model is for a one-dimensional system and is based on the assumption of a constant concentration at the source (C_0) and a zero concentration at the upper boundary (C=0 at z=L), which is assumed to be 1 m above the land surface:[12]

$$C(z,t) = C_0\left[\frac{e^{a(L-z)}-e^{-a(L-z)}}{e^{aL}-e^{-aL}} + \left(\frac{2\pi D_s}{L^2}\right)\sum_{n=1}^{\infty}(-1)^n\left(\frac{ne^{-t(\lambda+D_s\frac{n^2\pi^2}{L^2})}}{\lambda+D_s\frac{n^2\pi^2}{L^2}}\right)\sin\left(\frac{n\pi(L-z)}{L}\right)\right]$$

This equation was used to evaluate radon concentrations and fluxes for the following parameter values:

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[11]Based on additional studies, it is our belief that upward advection may be significant over long time periods. Estimates of the long-term average upward advection rate using water balance and isotope profile methods indicate the upward flux is on the order of hundredths to tenths of a millimeter per year (Chapman, 1996; Conrad and Strong, 1994).

[12]The equation for the analytical solution was presented incorrectly in the original report and is presented correctly in this final report. Rn flux rates were calculated correctly in the original report, but the analytical solution was incorrectly presented in the draft.

A fixed concentration of $C_0 = 4.345 \times 10^{-5}$ Ci/m^3 was assumed at the top of the waste, corresponding to the calculated K65 pore-space concentration for times less than 3700 days;

The land surface is 21.336 m above the top of the waste, and the zero-concentration condition is imposed 1 m above the land surface at L = 22.336 m;

The effective ^{222}Rn gas diffusion coefficient (D_S) is 126 m^2/yr (0.3456 m^2/d) (Baer et al., 1994);

The Rn-222 decay constant (λ) is 66.95 y^{-1} (0.1833 d^{-1});

$$a = \left(\frac{\lambda}{D_S} \right)^{\frac{1}{2}}$$

The time constant is given by:

$$\tau = \frac{1}{\lambda + \frac{\pi^2 D_S}{L^2}} = 1.44 \times 10^{-2}\ y = 5.26d$$

The source concentration is simulated using a step function based on the ^{222}Rn source concentration model results, presented in Section 5.5.1. This is a reasonable assumption given the relatively fast travel times for the gas phase diffusion and the relatively slow changes in the source concentration. This model was implemented using a spreadsheet program written for this evaluation. The input parameters and source step function can be seen in the spreadsheet output listed in Appendix 3.

Flux at the land surface (F) was calculated by multiplying the concentration gradient at the land surface (obtained by evaluating the derivative of the above equation with respect to z at z = 21.336 m) by the diffusion coefficient for radon in air (D_a = 347 m^2/yr = 1.1×10^{-5} m^2/s.) (Rogers and Nielson, 1991):

$$F = -D_a \, (dC/dz)$$

The assumptions for the gas phase diffusion model include:

1. stepwise steady state source concentration

2. zero concentration in the atmosphere at the upper boundary (3.3 ft (1 m) above the ground surface)

3. initial conditions are zero concentration above the source

4. barometric pumping is negligible

5. flux at the surface is a function of D_a and the concentration gradient in air.

In addition to these assumptions, the gas diffusion model does not account for the production of ^{210}Pb, along the transport path, as a result of the decay of ^{222}Rn.

Liquid phase diffusion model

The liquid phase diffusion of the modeled isotopes (Figure 3) is simulated using the SWIFT-II computer code (Reeves et al., 1986), in the same manner presented in Baer et al. (1994). For liquid phase diffusion in unsaturated sediments, the advective velocity is set to zero and the effective diffusion coefficient is a function of the molecular diffusion coefficient in water (D_m), volumetric water content (θ), sorption coefficient (K_d) and tortuosity (τ). The governing equation for one dimensional diffusive transport of a decaying, sorbing isotope is:

$$\frac{\partial C_i}{\partial t} = D_e \frac{\partial^2 C_i}{\partial z^2} + \lambda_{i-1} \frac{R_{i-1}}{R_i} C_{i-1} - \lambda_i C_i$$

Where:

$$R_i = 1 + \frac{\rho_b K_{d,i}}{\theta}$$

$$D_e = \frac{D_m}{\tau R_i}$$

Since this performance evaluation is based on a deterministic model, a single value must be selected for each parameter in the model. Some of these parameter values vary over time, and in space at the site, and a single representative value is not known, but a reasonable range of values for that parameter (for the existing hydraulic conditions) can be obtained from the data (e.g., water content, effective diffusion coefficient). For other parameters, the representative value is uncertain because there are limited or no data on that parameter for this site (e.g., sorption coefficient, tortuosity), and the range of possible values is limited only by the measured values for other sites and sediments.

Due to the time constraints imposed on this project, a set of "representative" parameter values could not be determined. Consequently, the mean values of the parameters

from the distributions used in the second GCD PA (Baer et al., 1994) were used. The only exception is the value chosen for the tortuosity. The smallest tortuosity was used because it minimizes the length of the simulation period. The input parameter values for the liquid diffusion model are listed in Table 5. To simplify the model it is assumed that all of the isotopes have the same molecular diffusion coefficient and distribution coefficient. The individual isotopes will have different source concentrations (see section 5.1.2) and decay coefficients. The SWIFT-II input files for the liquid diffusion model are located in Appendix 4.

Table 5. Liquid Diffusion Model Input Parameter Values

Parameter	Value
sediment porosity	0.35
water content	0.095
distribution coef.	0.003 ft^3 /lb
tortuosity	3
molecular diffusion coef.	0.000929 ft^2/d

Dose Models

The dose to an individual is estimated based on the concentration of the individual isotopes (C_i), the dose conversion factor (DCF_i) for a particular isotope (i) and exposure pathway (A) (e.g., inhalation of ^{222}Rn inside a home) and an exposure factor (f_A) which represents the relative length of time or amount of exposure from that pathway.

$$Dose_A = \sum_{i=1}^{m} C_i DCF_i f_A$$

Two sets of doses were calculated using the GENII code: (1) the dose to the intruder and MOP from isotopes subject to liquid diffusion, and (2) the dose to the intruder from the well drill cuttings and basement excavation tailings. The dose to the intruder from inhalation of radon was calculated using the dose conversion factor model above with dose conversion factors based on empirical models taken from Martin Marietta Energy Co. et al., (1994) and Baer et al., (1994).

The inhalation DCF for radon concentration in air to dose by inhalation is 360 mrem per year per pCi/l of radon in air (from Baer et al. (1994)). The dose conversion factor for radon inhalation based on radium concentration in soil is taken from Martin Marietta Energy Co. et al., (1994). The DCF for inhalation of radon while indoors is 240 mrem per year per 1 million pCi of ^{226}Ra per cubic meter of soil. The DCF while outdoors is equal to 0.28 of the indoor DCF. In these models the soil concentration includes the liquid phase and the sorbed phase. The doses calculated by GENII are also based on the bulk soil concentration (liquid and sorbed concentrations). The exposure factor for

radon inhalation is proportional to the relative amount of time spent breathing air with a given concentration of contaminants (radon).

Member of Public Dose Calculations

The MOP dose calculations are based on the following assumptions and parameter values:

1. The exposure pathways are external (from the ground surface) and internal (from inhalation)

2. Plants transport the waste from the root zone to the ground surface

3. 99% of native plant roots are in the upper 0.5 ft (0.15 m) of the sediments and the remaining 1% are at the maximum rooting depth of 14 ft (4.3 m)

4. The contaminated source area (for external exposure) is 78.5 ft^2 (7.3 m^2), the cross-sectional area of a GCD borehole

5. 1800 hours/yr are spent onsite and outdoors, with 100 of those hours spent landscaping (i.e., stirring up dust, but not gardening)

6. The resuspension of particulate matter is estimated using a mass loading model with a resuspension factor of 100 $\mu g/m^3$

The liquid phase concentration at the ground surface and at the maximum rooting depth at 100, 10,000 and 1 million years after closure are taken from the liquid diffusion simulation output file. The bulk soil concentration is then calculated by hand and used as input for the GENII code.

Intruder - Chronic Dose Model

The intruder's chronic dose calculations are based on the following assumptions and parameter values:

1. Exposure pathways are external (from the ground, basement tailings, and drill cuttings piles) and internal (inhalation of suspended particles and radon gas and ingestion of contaminated crops)

2. The excavated basement is 10 ft × 33 ft × 33 ft (3 m × 10 m × 10 m), floor is a concrete slab

3. The tailings from the basement are placed in a cone-shaped pile with a surface area of 2077 ft^2 (193 m^2)

4. 1800 hours/yr are spent outside, 100 of those hours are spent gardening

5. The same mass loading model used in the MOP scenario applies to the intruder

6. The intruder drills a 12 in. (30 cm) diameter well through one of the GCD boreholes, 750 ft deep to the water table

7. The drill cuttings from the well are evenly mixed and placed in a cone-shaped pile with a surface area of 296 ft^2 (27.5 m^2)

8. The contaminated source area for external exposure while gardening is 78.5 ft^2 (7.3 m^2) the cross-sectional area of a GCD borehole

9. 410 mg of soil is ingested per day (inadvertent)

10. Crops are grown on an area of 1615 ft^2 (150m^2)

11. Only 0.047 of that area is contaminated (a scaling factor for the dose calculation)

12. 4300 hours/yr are spent inside the home (at the site)

13. The indoor radon concentration is due to the radium in the soil at the basement elevation

14. 100 hours/yr are spent in close proximity to the basement tailings and well cuttings piles

15. The terrestrial food ingestion parameters shown in Table 6 apply to the garden scenario (these values represent 100% of the food ingested by the intruder)

The same methods are used in the calculation of the chronic dose to the intruder as were used in the calculation of the dose to the MOP. First, the liquid phase concentration at the ground surface and at the maximum rooting depth, at 100, 10,000 and 1 million years after burial, are taken from the liquid diffusion simulation output file. Then the bulk soil concentration is calculated by hand and used as input for the GENII code and for calculation of the radon doses. GENII is also used to calculate the decay of the isotopes in the cuttings pile from well drilling and to calculate the doses from the well cuttings pile after 100, 10,000 and 1 million years.

Intruder - Acute dose model

The intruder's acute dose calculations are based on the following assumptions and parameter values:

1. Exposure pathways are external (from the well cuttings and basement tailings) and internal (inhalation of suspended particles and radon gas)

Table 6. Terrestrial Food Ingestion Parameters

Crop	Growing Time (d)	Crop Yield (kg/m^2)	Production (kg/yr)	Holdup (d)	Consumption (kg/yr)
leafy vegetables	60	2.2	60	1	60
root vegetables	90	9.0	182	1	182
fruit	60	5.2	335	1	335
grain	90	2.1	88	1	88

2. The dimensions of the basement are 10 ft × 33 ft × 33 ft (3 m × 10 m × 10 m) and the area is 1089 ft^2 (100 m^2).

3. It requires 40 hours to excavate the basement

4. It takes four hours to drill through the 50 ft segment of wastes

5. The same mass loading model used in the MOP scenario applies to the intruder

6. The diameter of the water well borehole is 12 in. (30 cm)

7. The waste is evenly distributed in the borehole

The same methods are used in the calculation of the acute dose to the intruder as were used in the calculation of the chronic intruder dose. First, the liquid phase concentration at the basement elevation at 100, 10,000 and 1 million years after closure are taken from the liquid diffusion simulation output file. Then the bulk soil concentration is calculated by hand and used as input for the GENII code. GENII is used to calculate the decay of the isotopes in the well cuttings pile and to calculate the acute dose from the well cuttings and basement tailings piles at 100, 10,000 and 1 million years after burial.

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6. Results of Analysis

The calculated doses to the MOP and the intruder from the K-65 and Silo 3 wastes are summarized in Tables 7 and 8.[13] The radon fluxes at the ground surface, calculated using the gas diffusion model described in the previous section, are shown in Figures 11 and 12. A discussion of the type of dose (internal, external), the isotope that is most responsible for the dose, and a comparison with the performance objectives follows each of the tables.

Table 7. Calculated Doses from Exposure to K-65 Wastes (EDE)

Scenario	100 yr (mrem)	10,000 yr (mrem)	1,000,000 yr (mrem)
MOP	0	8.9	13
INTRUDER - CHRONIC			
Basement Tailings	0	18	13
Well Cuttings	720	220	11
Garden	0	85	137
Radon Inhalation (all airborne pathways)	10504	194	1276
TOTAL CHRONIC	11,224	516	1,437
INTRUDER - ACUTE			
Basement Tailings	0	0.4	0.6
Well Cuttings	94	20	0.6
TOTAL ACUTE	94	20.4	1.2

The dose to the MOP increases over time as the contaminants diffuse toward the ground surface. After 10,000 years the dose to the MOP is 8.9 mrem, increasing to 13 mrem after 1 million years. The dose at 1 million years is the maximum dose the MOP would receive under the modeled conditions. All of the dose to the MOP is internal from the inhalation and consumption of contaminated soil particles (i.e., EDE

[13]As noted in Section 3.5, doses presented in Tables 7 and 8 were calculated as EDE, which includes both internal and external exposures. However, the DOE standard for intruder exposures is specified as CEDE, internal exposures only. We have tried to clarify which doses are CEDE and which are EDE.

Table 8. Calculated Doses from Exposure to Silo 3 Wastes (EDE)

Scenario	100 yr (mrem /yr)	10,000 yr (mrem/yr)	1,000,000 yr (mrem/yr)
MOP	0	2.2	14
INTRUDER - CHRONIC			
Basement	0	4	15
Well	190	330	0
Garden	0	21	141
Radon inhalation	2	4	133
TOTAL CHRONIC	192	359	289
INTRUDER - ACUTE			
Basement	0	0.1	0.4
Well	8	31	0.5
TOTAL ACUTE	8	31	1

approximately equal to CEDE). The isotope that is responsible for most of this dose is ^{227}Ac. Importantly, the solubility of actinium is the highest (along with protactinium) of all the elements.

Recall that this value was selected arbitrarily, as a conservative estimate of the solubility. The solubility of actinium and protactinium could not be modeled for the expected chemical conditions due to a lack of thermodynamic data on these elements.

The total chronic dose that the intruder receives decreases between 100 and 10,000 years, then increases again between 10,000 and 1 million years. At 100 years, almost all of the intruder's dose is from the well cuttings pile and the most of that dose is internal due to the inhalation of radon from the cuttings pile. This dose is very high (11 rem) due to the high initial concentration of ^{226}Ra in the K-65 wastes. As the ^{226}Ra decays, the dose due to the well cuttings decreases. By 10,000 years the chronic dose to the intruder has decreased significantly. The majority of the dose is still due to the well cuttings pile, but a smaller percentage is due to the inhalation of ^{222}Rn from the well cuttings pile.

By 10,000 years, a significant portion of the dose is due to the liquid phase diffusion of contaminants up to the accessible environment, as evidenced by the increased dose due to gardening and from the basement tailings pile. The isotope contributing most to the gardening dose is ^{231}Pa, with ^{227}Ac controlling the dose from the basement tailings pile. As mentioned previously, these elements have the highest solubilities in the

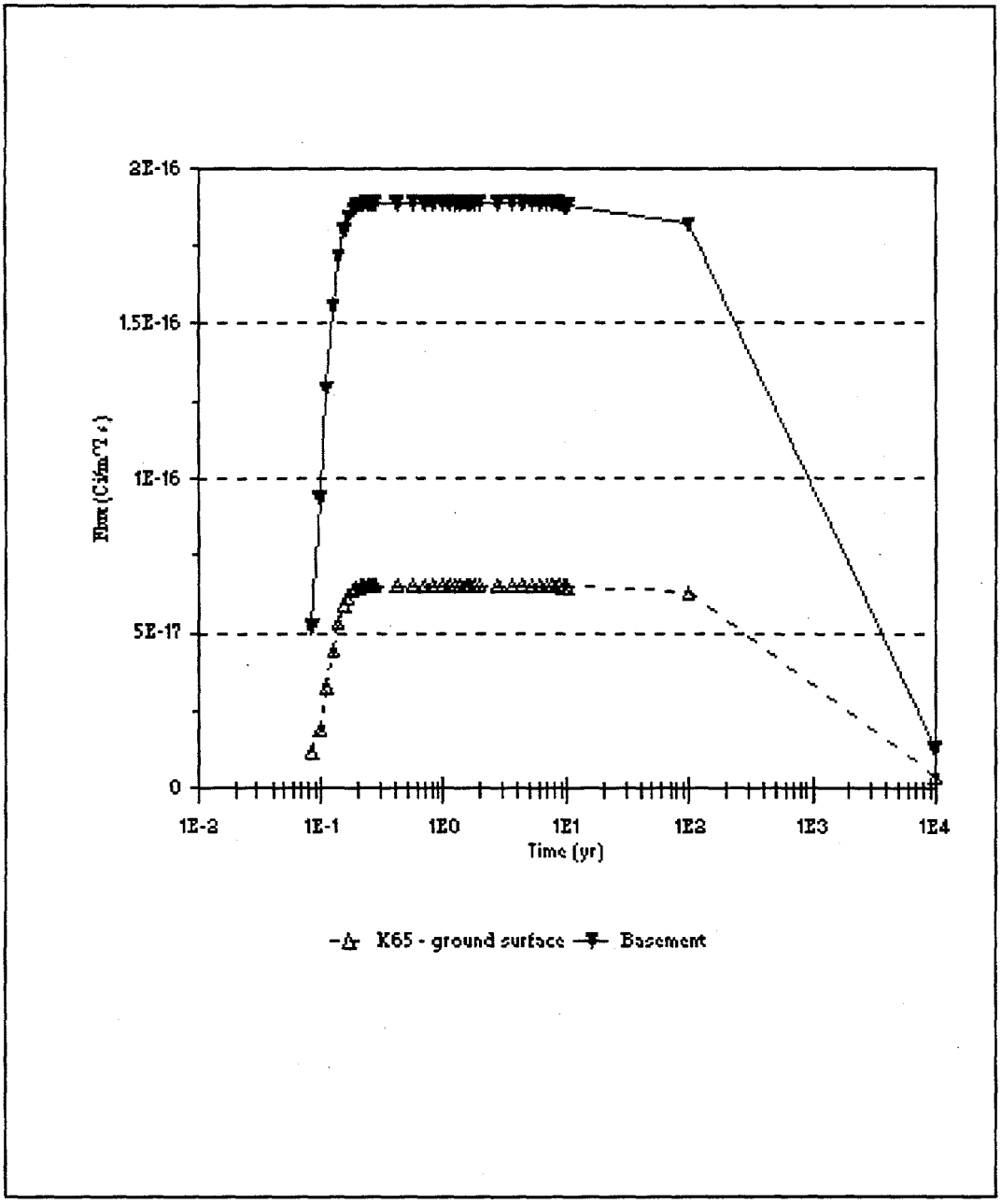

Figure 11. Radon flux ($Ci/m^2/s$) at the ground surface and basement: K-65 wastes.

model. By 1 million years, the intruder's chronic dose has increased dramatically. Once again, almost all of the dose is due to the inhalation of ^{222}Rn. However, this time the ^{222}Rn is present at the accessible environment due to upward movement by liquid phase diffusion and subsequent decay of ^{226}Ra to ^{222}Rn.

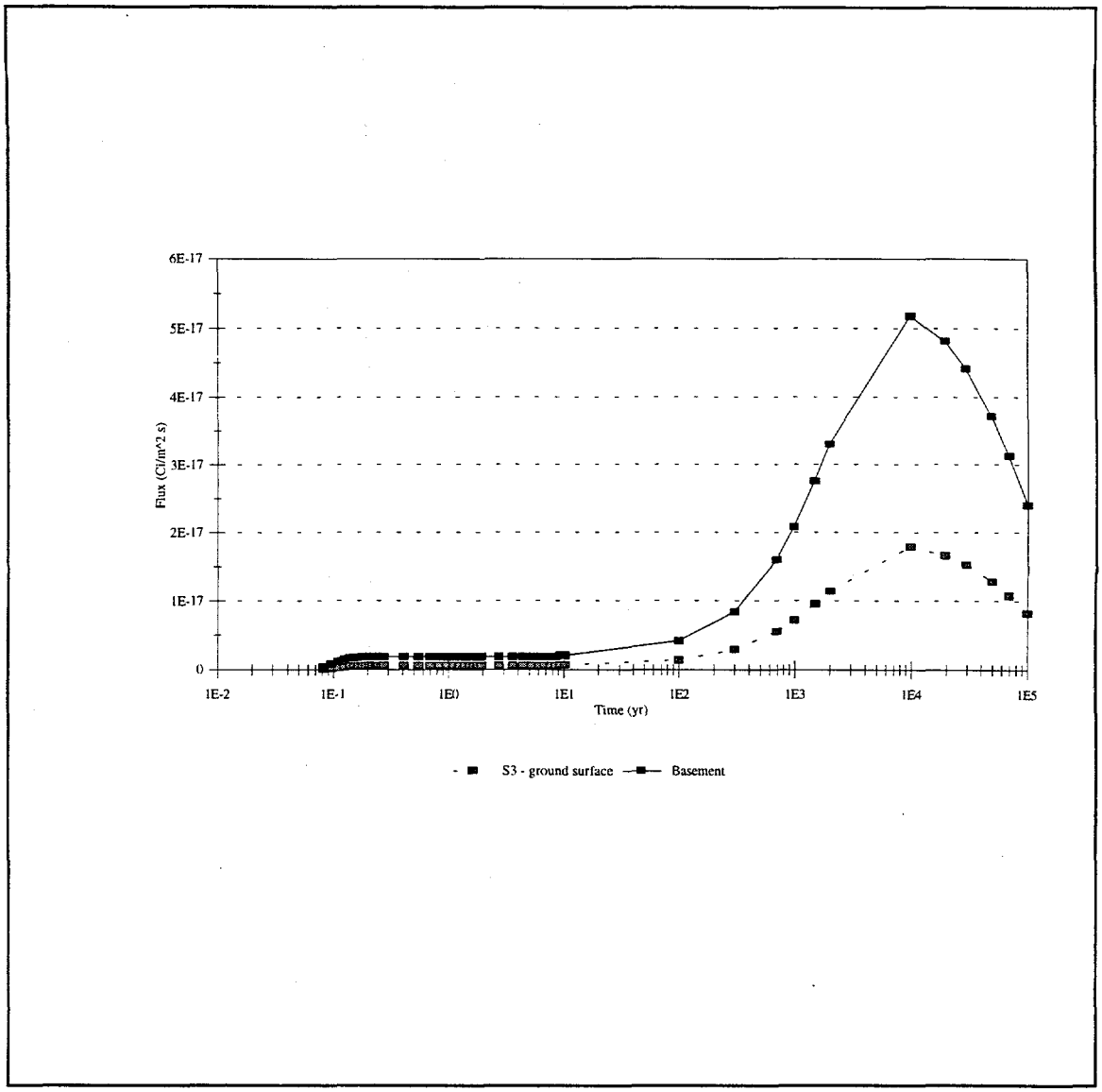

Figure 12. Radon flux ($Ci/m^2/s$) for Silo 3 wastes.

The acute dose to the intruder is small relative to the chronic dose, because of the shorter exposure times. The dose due to excavating the basement increases slightly over time as the contaminants diffuse to the surface in the liquid phase. As in the chronic dose, ^{227}Ac is the primary contributor to the basement scenario dose. Most of the acute dose is due to drilling a well through the wastes. The acute dose decreases over time as the ^{226}Ra concentration decreases.

K-65 Wastes vs. Performance Objectives

The results of the dose models for the Silo 3 wastes are similar to the K-65 results for the scenarios that involve liquid phase diffusion of the contaminants to the ground surface (e.g., MOP, garden, and basement scenarios). The results for the two wastes differ when the dose due to inhalation and external exposure to ^{222}Rn are added. The

doses as a result of inhalation of radon are much smaller for the Silo 3 wastes than for the K-65 wastes for all three simulation times because the Silo 3 wastes contain about 4,000 pCi/g of ^{226}Ra and the K-65 wastes contain about 400,000 pCi/g of ^{226}Ra.

The intruder's acute and chronic doses are highest at 10,000 years. This coincides with the modeled, peak ^{222}Rn concentration in these wastes.

If the inhalation of contaminated soil particles is classified as an exposure due to the soil pathway, then the K-65 wastes meet the performance objective for MOP (i.e., less than 25 mrem/yr EDE) for all three simulations. However, if this pathway is considered an air pathway, the maximum dose of 13 mrem per year (at 1,000,000 yr) exceeds the MOP performance objective of 10 mrem per year for the air pathway. The major portion of the dose, in this case, is due to ^{227}Ac. The solubility for this element is modeled using a very large value and it is likely that, given a better estimate of the solubility of actinium for the modeled conditions, the dose would not exceed either of the MOP performance objectives.

The acute dose to an intruder meets the performance objective for all three simulations (i.e., less than 500 mrem CEDE). The chronic intruder-dose exceeds the performance objective of 100 mrem CEDE for all the simulations. For this PE, doses to the intruder were actually calculated as EDE and includes external exposure as well as internal exposure; however, almost all the dose is internal and CEDE ≈ EDE.

Silo 3 Wastes vs. Performance Objectives

As with the K-65 wastes, the Silo 3 wastes meet the MOP performance objectives, for the modeled conditions, if the inhalation of soil particles is considered a soil pathway. Otherwise, these wastes exceed the air pathway performance objective at the time of the peak MOP dose (1,000,000 yr). This high dose is probably a function of the conservative value assigned for the solubility of actinium and protactinium. The Silo 3 wastes meet the performance objective of 500 mrem CEDE for the acute dose to the intruder. However, these wastes exceed the performance objective of 100 mrem CEDE for the chronic dose to the intruder, at all three simulation times. This is because most of the EDE dose is from inhalation of radon.

Radon Fluxes vs. Performance Objective

The maximum radon fluxes shown for K-65 waste (Figure 11) and the Silo 3 waste (Figure 12) are approximately four orders of magnitude below the performance objective of 20 $pCi/m^2/s$.

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7. Assumptions, Simplifications And Qualifiers

This section summarizes major assumptions and simplifications used in the analyses. Most of these assumptions and simplifications may not be appropriate for a formal, 5820.2A PA; however, they were necessary so that a preliminary evaluation could be completed in the allotted six weeks. The list of assumptions and simplifications follows.

1. For this preliminary evaluation, only the dose from Fernald byproduct material was calculated; we did not integrate this Fernald dose with the doses from the other LLW at the Area 5 RWMS.

2. Occupational radiological doses to workers at the RWMS were not assessed; however, the radon flux at the land surface was calculated for the entire time period.

3. Performance was assessed only for the post-closure period.

4. Contaminated sediments were not transported by surface water.

5. Surface water pathway was not significant; that is, we assumed there is not, and will never be, any use of surface water.

6. Upward advection of radionuclides was negligible.

7. Barometric pumping of the gas phase was assumed not to occur.

8. For the undisturbed case, the depth of burial was assumed to remain constant through time:

   a. there is no subsidence of the boreholes

   b. there is no surface erosion

   c. there is no cap on the boreholes

9. Mean input parameter values from the distributions used for the second iteration of the GCD PA were assumed for modeling. These values are not necessarily representative or conservative.

10. Current versions of DOE Order 5820.2A will remain in effect until a disposal decision is made for this byproduct material.

11. Passive controls (institutional knowledge and engineered barriers) do not exist.

12. The probability that the inadvertent intruder would drill a well through a GCD borehole is one.

13. Wastes from silos 1 and 2 will be kept separate from silo 3 wastes and the volume of silos 1 and 2 equals 64% of the total volume.

14. Uniform waste characteristics for all Silo 1, 2, and 3 wastes.

15. The inadvertent intruder does not contribute to the MOP dose.

16. The regulatory drivers and performance objectives defined in this report are correct.

17. Climate will be static.

18. Temporal evolution of the system (e.g., geologic changes) were not analyzed or incorporated into the analysis. That is, the system was modeled to remain as it stands today.

19. No significant recharge to groundwater will occur; consequently, the groundwater pathway was not evaluated.

20. The analysis was deterministic, not probabilistic. There was no evaluation of effects of uncertainty in the parameter values on model results.

21. There is no quality assurance beyond what is contained in this document.[14]

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[14]Assumptions 22 and 23, of the original report, duplicated assumptions 10 and 11 of the original report, therefore, the original assumptions 22 and 23 were deleted.

8. Outstanding Issues

In this section, issues which may require DOE resolution are identified, and solutions to those issues are recommended.

Issue 1 - The burial of this vitrified byproduct waste may be defined as a significant federal action, triggering a NEPA evaluation.

Recommendation 1 - If the current site-wide NEPA EIS for the NTS is sufficiently broad to cover the disposal of this waste using the GCD concept, no significant NEPA actions would be required. If the site-wide EIS is not sufficiently broad, a separate NEPA action will be necessary. Such action would probably involve the development and approval of a separate NEPA Environmental Assessment or EIS. Sufficient lead time and funds may need to be allocated for the completion of the NEPA process.

Issue 2 - On June 29, 1992 the U.S. EPA granted Nevada authority to regulate mixed wastes; that is, wastes containing both radioactive and hazardous components as defined by the Atomic Energy Act and the RCRA, respectively. The State may want to use this authority to regulate the disposal of these silo wastes. Prior to vitrification, these wastes meet the definition of a RCRA hazardous waste because they failed the EPA's TCLP test.

Recommendation 2 - Bring this issue to the attention of the State of Nevada and request a formal statement of their position. The vitrified wastes pass the TCLP test and no longer qualify as a RCRA hazardous waste. Additionally, byproduct is specifically excluded from RCRA by the Byproduct Rule (10 CFR 962).

Issue 3 - The State of Nevada has taken the position that the GCD boreholes are Class IV injection wells and the disposal of solid radioactive wastes is injection and violates UIC regulations.

Recommendation 3 - Bring this issue to the attention of the State of Nevada and request a formal statement of their position. On the Federal side, for TRU radioactive waste disposal at the WIPP, the EPA has taken the position the disposal of 40 CFR 191 wastes does not constitute underground injection. The EPA position is that compliance with 40 CFR 191 provides equivalent protection to the UIC regulations. This EPA position may provide an important basis for countering the State's definition of underground injection. If the state maintains that this is underground injection, there are three alternatives: 1. settle the issue in court (or wait for Yucca Mountain to settle the issue in court), 2. use "brute force" GCD disposal (i.e., very large trenches), or 3. do not use the GCD concept.

Issue 4 - The FEMP "OU4 NTS Application, Draft, December 7, 1993" notes that the "... highest TRU concentrations were associated with some materials ... determined through analysis to contain in one case up to 93,904 dpm/g (42 nCi/g)."

Recommendation 4 - Present additional information to clarify the concentration of TRU radionuclides in these wastes. We suspect that a number of additional analyses exist and that this issue has been resolved; however, citing one analysis was insufficient. This issue is important because wastes containing 100 nCi/g or more of TRU wastes must be disposed under the requirements of 40 CFR 191.[15]

Issue 5 - Need to determine if the inhalation of contaminated soil particles is an air pathway or soil pathway, in order to establish the performance objectives for the MOP.

Recommendation 5 - Request a clarification of 40 CFR 61.92 to determine if this rule applies to resuspended soil particles.

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[15]The original report states that the 100 nCi/g limit is important because 100 nCi/g of transuranics violates the NRC's Class C limits; the correct concern, for DOE wastes, is that 100 nCi/g would require compliance with 40 CFR 191.

9. Activities to Complete a Performance Assessment

This preliminary evaluation establishes a strong foundation for the completion of a full PA. The following activities would need to be completed to convert this evaluation to a full PA for submittal to the DOE Peer Review Panel.

Item 1 - Order 5820.2A states that the inadvertent intruder cannot receive more than 100 mrem chronic dose, and no more than 500 mrem acute dose from all pathways, including radon.

Other performance assessments and the "guidance documents" by Wood et al. (1992 and 1994) assign a probability of one to the intruder scenario, that is, the consequence of disturbing the waste is estimated without an evaluation of the probability that the waste would be disturbed. For this preliminary evaluation, we have estimated the consequence of drilling through the waste without evaluating the probability of such an event. Ignoring probability places all sites on equal footing. For a given waste configuration, the intruder receives the same dose independent of site setting. Use of the GCD concept at Fernald Ohio will result in the same dose to the intruder as use of the GCD concept at the NTS!

Intuitively, the probability that someone would drill through a GCD borehole is much higher at Fernald (higher population density, shallower groundwater) than at NTS.

For future work, we propose to evaluate risk to the inadvertent intruder (i.e., probability times consequence). DOE Order 5820.2A does not preclude an evaluation of risk and the regulation (40 CFR 191) that governs the disposal of high level, spent nuclear fuel and TRU radioactive wastes provides a framework for including probability in the evaluation of disruptive scenarios (such as inadvertent intrusion). Both 40 CFR 191 and 40 CFR 194 offer guidance on rates of exploratory drilling per square kilometer per 10,000 years. The guidance on rates provided in 40 CFR 191 and 40 CFR 194, along with information about potential resource exploration at a particular site, could be applied to a LLW PA. Such an analysis will result in a more realistic evaluation of the suitability of this disposal concept at the NTS.[16]

Item 2 - This preliminary evaluation was deterministic; that is, single values were used for all input parameters, such as moisture content. There are a number of shortcomings to a deterministic evaluation. First, many parameters have a range of values; for example, the actual soil moisture content is a range of values, not a single value. Second, there is uncertainty in many input parameters; for example, the

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[16]DOE/NV has taken a number of steps to implement this recommendation. As part of the "white paper" on "A Proposed Approach for Conducting Performance Assessment at the Nevada Test Site," DOE/NV funded the development of the concept of assessing the probability of inadvertent human intrusion, for LLW wastes that fail the inadvertent human intruder standard, *independent of the depth of burial.* The full development of this recommendation is documented in Cochran, J.R., 1996. DOE/NV then convened an expert elicitation to assess the probability of inadvertent human intrusion by water well drilling. The elicitation process and its results are documented in Black, P. et al., 1997.

solubility of ^{226}Ra varies with the decay rate of the glass and the pH of the pore water, and a single-valued solubility cannot capture our uncertainty in the chemistry. Third, a PA involves modeling a number of processes and a unique combination of values may be very important to the outcome (e.g., high water content, coupled with high pH). Finally, by running the model many times, and sampling from a range of values each time, a probability can be assigned to various outcomes. A probabilistic PA provides the decision maker with a much more robust analysis on which to base a determination.

We recommend a probabilistic analysis to support future PA work for this waste. We have completed two such analyses of the TRU wastes in the existing GCD boreholes; this experience could be easily applied to a LLW PA. The decision maker can observe explicitly the uncertainty in the output, and can then use the mean or median of the output, as well as the range, for decision making. A deterministic analysis does not allow this.[17]

Item 3 - The Fernald waste was evaluated as presented, 3253 concrete boxes containing 846,504 three gallon cans of vitrified waste. About 12% of the bottom 50 ft section of a GCD borehole contains actual waste. Other alternatives should be evaluated, such as placing the vitrified wastes directly in the borehole without the concrete boxes or the three-gallon cans; borehole utilization would jump from 12% to 70%. Tradeoffs would have to be evaluated because the inadvertent intruder would drill through much more "concentrated" source term. On the other hand, due to the smaller number of boreholes, the probability of inadvertently drilling into or excavating one of the boreholes would accordingly decrease.

Item 4 - This preliminary evaluation was based on the current GCD borehole configuration, 120 ft deep, 10 ft in diameter, with 70 ft of backfill. A better borehole configuration might be derived from boreholes 9 ft in diameter, 150 ft deep, with 72 ft of backfill. We propose to optimize the borehole configuration. Tradeoffs would have to be evaluated between costs of various configurations and performance (especially performance under the inadvertent intruder scenario).

Item 5 - For this preliminary evaluation, no credit was taken for passive controls. The loss of active institutional controls at 100 years is not the same as the loss of all controls at 100 years.[18] The cost/benefit of installing passive controls (e.g., markers and engineered barriers) should be explored.

Item 6 - The parent radionuclides in the Fernald silo wastes have very long half-lives and there is little change in the activity of these wastes for the first 100 million years. For this preliminary evaluation, climate change was not assessed; however, an ever-changing climate is generally recognized as the norm. Climate change could bring deep-rooted (70 m) plants to the GCD elevation. As recently at 10,000 years ago

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[17]DOE/NV has implemented this recommendation and the probabilistic PA for these OU4 wastes is presented by Brown, T.J. et al., draft 1997.

[18]DOE/NV has used the expert elicitation process to evaluate the effectiveness of active and passive controls, as documented by Black, P., et al., 1997.

juniper trees grew at the GCD elevation at the NTS (Wigand et al. in review, 1995). Additionally, there is no significant groundwater recharge under current conditions (Conrad 1993); however, this could change with significant climate change over longer time scales.

We are currently assessing climate change and its effects as part of our 40 CFR 191 PA for the TRU in the GCD boreholes. Results from the 40 CFR 191 work should be incorporated into a PA for the Fernald byproduct materials because of the long-lived nature of many of the radionuclides.

Item 7 - For this preliminary evaluation, only the dose from the Fernald byproduct material was calculated. The Fernald dose should be integrated with the doses from the other LLW wastes buried at the Area 5 RWMS after FY 1987.

Item 8 - In this analysis, important surface processes were assumed to be negligible. Based on experience gained in conducting the GCD 40 CFR 191 analyses, the following surface processes should be analyzed to determine if they are truly negligible. First, contaminated sediments could possibly be transported by surface waters. Second, subsidence of the boreholes may occur and have an impact on the migration of radionuclides. Third, surface erosion could alter the depth of burial, enhancing the movement of radionuclides. Each of these surface processes should be evaluated and either defensibly dismissed, or incorporated into the modeling.[19]

Item 9 - The upward advection of pore water was neglected; however, this may be an important transport mechanism. Consequently, its importance should be evaluated.

Item 10 - Our analysis of regulatory drivers and the extraction of performance objectives was preliminary, and a more through analysis needs to be conducted.[20]

Item 11 - Under current conditions, there is not believed to be significant recharge to groundwater. In a full PA, this assumption needs to be better documented and defended.

Item 12 - A much more robust and complete QA program will be required in order for the PA analyses and supporting evidence to be defensible.

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[19]The full performance assessment (Brown, T.J. et al., draft 1997) provides a more robust treatment of surface processes.

[20]A more thorough review of regulatory drivers is presented in Brown, T.J. draft 1997.

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10. Benefits of GCD Disposal

In FY93, Sandia developed a white paper for the DOE that explained potential advantages of the GCD concept for disposal of special-case waste. As this paper serves as a concise, effective summary, it has been included as an attachment to this report (Attachment A). Although a second PA iteration has been completed and a third performance assessment iteration has been initiated since the writing of this white paper, almost all of the information in the paper still applies. The draft of the white paper appended is noted as a pre-decisional draft; however, this draft has since been approved by DOE/NV and DOE/HQ.

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Appendices

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Appendix 1: Summary of Radionuclide Analyses for Silo 1 and 2 Byproduct

[Remedial Investigation Report for Operable Unit 4, Fernald Environmental Management Project, Fernald Ohio, Volume 1 of 3, 1993.]

Analyte	Frequency of Detection[a]	Arithmetic Mean[b] (pCi/g)[c]	Upper 95% CI on A Mean[b] (pCi/g)	Range of Detection[b] (pCi/g)
SILO 1				
Actinium-227	13/20	5960	7670	4320-17390
Lead-210	20/20	165000	202000	48980-381400
Polonium-210	13/13	242000	281000	144000-434000
Radium-226	20/20	391000	477000	89280-890700
Thorium-228	2/20	422	2280	835-2280
Thorium-230	24/24	60000	68900	10569-105372
Thorium-232	8/20	424	1110	661-1106
Uranium-234	21/21	800	932	326-1548
Uranium-235/236	14/20	38	54	19.1-105
Uranium-238	20/20	642	693	387-920
SILO 2				
Actinium-227	11/14	5100	6640	2905-10450
Protactinium-231	1/14	2350	4040	4041-4041
Lead-210	14/14	145000	190000	58160-399200
Polonium-210	8/8	139000	231000	55300-241000
Radium-226	14/14	195000	263000	657-481000
Thorium-228	5/14	645	7360	411-7360
Thorium-230	15/15	484000	76200	8365-132800

Thorium-232	3/14	402	985	851-985
Uranium-234	13/13	961	1160	121-1465
Uranium 235/236	11/13	73	94	35.6-172
Uranium 238	14/14	912	1120	46-1925

[a]Rejected data not included in total number of samples.
[b]Values qualified with an R are excluded. The mean and upper 95% confidence interval (CI) on mean have been rounded to show three significant figures. The mean is calculated using one-half the SQL for nondetects.
[c]Values expressed in picoCuries per gram (pCi/g).

RADIONUCLIDE CONCENTRATIONS IN SILO 3 BYPRODUCT

Analyte[a]	Frequency of Detection[b]	Arithmetic Mean[c] (pCi/g)[d]	Upper 95% CI on A Mean[c] (pCi/g)	Range of Detection[c] (pCi/g)
SILO 3				
Actinium-227	9/9	618	925	234-1363
Lead-210	11/11	2620	3480	454-6427
Protactinium-231	9/11	487	627	266-931
Radium-224	11/11	290	367	64-453
Radium-226	11/11	2970	3870	467-6435
Radium-228	9/11	297	406	82-559
Thorium-228	7/11	590	747	459-996
Thorium-230	11/11	51200	60200	21,010-71,650
Thorium-232	8/11	656	842	411-1451
Uranium-234	11/11	1480	1730	348-1935
Uranium-235/236	10/11	93.6	117	42-158
Uranium-238	11/11	1500	1780	320-2043

[a]Sample numbers used in this data set include: 100097 through 100107.
[b]Rejected data not included in total number of samples.
[c]Values qualified with an R are excluded. The mean and upper 95% confidence interval (CI) on mean have been rounded to show three significant figures. The mean is calculated using one-half the SQL for nondetects.
[d]Values expressed in picoCuries per gram (pCi/g).

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Appendix 2: Spreadsheet Output for Source Term Calculations: Mass in Borehole and Isotope Solubilities

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			K65 waste	note: 0.741 Vsolid/Vwaste; Vborehole=3927 ft³ 208.2 bh Ci waste/borehole			
ID	ATWT (G/MOL)	T1/2(YR)	CONC(pCi/in³SOLID)	CONC(Lb/in³ SOLID)	CONC(LB/FT³SOLID)	LB/ft³borehole	
U238	238	4.4700E+09	51111	0.00033548232450117	0.57971345673802	9.079E-02	0.054319532697694
TH234	234	6.6000E-02	51111	4.8701793653263E-15	8.4156699432839E-12	1.318E-12	0.054319532697694
PA234	234	7.6400E-04	51111	5.6376015683474E-17	9.7417755101043E-14	1.526E-14	0.054319532697694
U234	234	2.4600E+05	52946	1.8804201877445E-08	3.2493660844226E-05	5.089E-06	0.056269726247034
TH230	230	7.5400E+04	3477876	3.7212089833767E-07	0.0006430249123275	1.007E-04	3.6962023654502
RA226	226	1.6000E+03	21770952	4.8570953283006E-08	8.3930607273034E-05	1.314E-05	23.137640410556
RN222	222	1.0000E-02	21770952	2.9819556495651E-13	5.1528193624484E-10	8.070E-11	23.137640410556
PB210	210	2.2300E+01	9219559	2.6638213224135E-10	4.6030832451305E-07	7.209E-08	9.7983239725073
U235	235	7.0400E+08	4293	4.381998202655E-06	0.0075720928941878	1.186E-03	0.004562496407255
TH231	231	3.0000E-03	4293	1.8355444598887E-17	3.1718208266877E-14	4.968E-15	0.004562496407255
PA231	231	3.2800E+04	184387	8.6195959247429E-09	1.4894661757956E-05	2.333E-06	0.1959620370474
AC227	227	2.1770E+01	350077	1.0673789901096E-11	1.8444308949093E-08	2.889E-09	0.37205335540707
TH227	227	5.1000E-02	345177	2.4655207865716E-14	4.2604199191957E-11	6.673E-12	0.36684575410366
RA223	223	3.1300E-02	730257	3.144819432817E-14	5.4342479799077E-11	8.511E-12	0.77609945000529
TH232	232	1.4000E+10	50669	0.0010153812487436	1.754578797829	2.748E-01	0.053849785804611
RA228	228	5.7600E+00	48391	3.9209627147125E-13	6.7754235710231E-10	1.061E-10	0.051428782586413
AC228	228	7.0400E-04	48391	4.7922877624263E-17	8.2810732534727E-14	1.297E-14	0.051428782586413
TH228	228	1.9130E+00	335918	9.0396970929838E-13	1.5620596576676E-09	2.446E-10	0.35700551319176
RA224	224	1.0000E-02	335918	4.6425017856319E-15	8.022243085572E-12	1.256E-12	0.35700551319176
PB212	212	1.2000E-03	335918	5.2725555993963E-16	9.1109760757568E-13	1.427E-13	0.35700551319176
PO210	210	3.7900E-01	12825237	6.297884464265E-12	1.088274435425E-08	1.704E-09	13.630351207708

ID	ATWT(G/MOL)	T1/2(YR)	K65 waste CONC(pCi/in³ SOLID)	TOTAL MASS (MOL)	168C/BOX MOL/BH	ion sol. mol/kg	g/g	SILO3 CONC(pCi/in³ Solid)	TOTAL MASS (MOL)	168C/BOX MOL/BH	ion sol mol/kg	g/g
AC227	227	2.1770E+01	350077	0.0038191238624448	1.834E-05			42122	0.000258482815567	2.207E-06		
AC228	228	7.0400E-04	48391	1.707178733568E-08	8.200E-11			18488	3.66882379776E-09	3.133E-11		
AC				0.0038191409342321	1.834E-05					2.2072E-06		
PA231	231	3.2800E+04	184387	3.030720440832	1.456E-02			45531	0.420964327584	3.595E-03		
PA234	234	7.6400E-04	51193	1.959953082624E-08	9.414E-11			122958	2.64797823946E-08	2.261E-10		
PA				3.0307204604315	1.456E-02					0.00359467		
PB210	210	2.2300E+01	9219559	0.10302835055558	4.949E-04	2.3E-19	4.82E-20	446283	0.002805301020492	2.395E-05	1.66E-19	3.49E-20
PB211	211	6.8683E-05	867187	2.9847388612603E-08	1.434E-10	6.65E-26	1.4E-26	27322	5.28967019589E-10	4.517E-12	3.14E-26	6.62E-27
PB212	212	1.2000E-03	335918	2.02002273792E-07	9.703E-10	4.5E-25	9.55E-26	18209	6.159303504E-09	5.260E-11	3.65E-25	7.75E-26
PB214	214	5.0989E-05	19990940	5.1080411730411E-07	2.454E-09	1.14E-24	2.44E-25	209481	3.01084458658E-09	2.571E-11	1.79E-25	3.82E-26
PB				3113650	1.212E+04					256		
RA223	223	3.1300E-02	730257	1.1454121939392E-05	5.502E-08	6.56E-15	1.46E-15	31879	2.81263443876E-07	2.402E-09	9.21E-20	2.05E-20
RA224	224	1.0000E-02	335918	1.6833522816E-06	8.086E-09	9.64E-16	2.16E-16	18209	5.13275292E-08	4.383E-10	1.68E-20	3.77E-21
RA226	226	1.6000E+03	21770952	17.455775145984	8.384E-02	1E-08	2.26E-09	214037	0.096532399296	8.243E-04	3.16E-14	7.14E-15
RA228	228	5.7600E+00	48391	0.0001396782600192	6.709E-07	8E-14	1.82E-14	18488	3.00176492544E-05	2.563E-07	9.83E-18	2.24E-18
RA				17.455927961718	8.385E-02					0.00082456		
TH227	227	5.1000E-02	345177	8.82173001024E-06	4.237E-08	2.48E-25	5.63E-26	41532	5.9705904816E-07	5.098E-09	3.94E-18	8.95E-19
TH228	228	1.9130E+00	335918	0.00032202529147008	1.547E-06	9.06E-24	2.06E-24	34026	1.83480591074E-05	1.567E-07	1.21E-16	2.76E-17
TH230	230	7.5400E+04	3477876	131.40962487245	6.312E-01	3.7E-18	8.5E-19	6466625	137.440044027	1.174E+00	9.08E-10	2.09E-10
TH231	231	3.0000E-03	4293	6.45392448E-09	3.100E-11	1.81E-28	4.19E-29	5327	4.50472428E-09	3.847E-11	2.97E-20	6.87E-21
TH232	232	1.4000E+10	50669	355477.48992	1.707E+03	1E-14	2.32E-15	38336	151286.12352	1.292E+03	9.99E-07	2.32E-07
TH234	234	6.6000E-02	51111	1.69044112512E-06	8.120E-09	4.75E-26	1.11E-26	122958	2.28752046864E-06	1.953E-08	1.51E-17	8.53E-18
TH				355608.89987742	1.708E+03					1293.02493		
U234	232	2.4600E+05	52946	6.52694568192	3.135E-02	3.16E-08	7.4E-09	78787	5.46328597176	4.665E-02	1.11E-10	2.59E-11
U235	235	7.0400E+08	4293	1514.52094464	7.275E+00	7.34E-06	1.73E-06	5327	1057.10863104	9.027E+00	2.14E-08	5.04E-09
U238	238	4.4700E+09	51111	114488.9671104	5.499E+02	0.000555	0.000132	122958	154927.5226488	1.323E+03	3.14E-06	7.47E-07
U				116010.01500072	5.572E+02					1332.01912		

FIXED CONC. LOWER BC AND ZERO CONC. UPPER BC, W/DECAY PA2 MODEL adapted for Rn222

K-65	initial conc.	time	n								
	Ci/m³	d	yr	1	2	3	4	5	6	C(x) [Ci/m³] flux	[Ci/m² s]
dist. to surface	4.3450E-05	30	0.083333	-0.00245	0.004741	-0.00324	0.001119	-0.00022	2.65E-05	1.0895E-12	1.1985E-17
m	4.3450E-05	35	0.097222	-0.00095	0.001654	-0.00095	0.000259	-3.8E-05	3.1E-06	1.7943E-12	1.9737E-17
21.336	4.3450E-05	40	0.111111	-0.00037	0.000577	-0.00028	5.99E-05	-6.4E-06	3.62E-07	2.9914E-12	3.2905E-17
	4.3450E-05	45	0.125	-0.00014	0.000201	-8.2E-05	1.38E-05	-1.1E-06	4.22E-08	4.0723E-12	4.4795E-17
effective D	4.3450E-05	50	0.138889	-5.5E-05	7.01E-05	-2.4E-05	3.21E-06	-1.9E-07	4.93E-09	4.8606E-12	5.3466E-17
m²/d	4.3450E-05	55	0.152778	-2.1E-05	2.45E-05	-7.1E-06	7.42E-07	-3.2E-08	5.76E-10	5.3603E-12	5.8963E-17
0.3456 SOIL	4.3450E-05	60	0.166667	-8.2E-06	8.53E-06	-2.1E-06	1.72E-07	-5.4E-09	6.73E-11	5.6465E-12	6.2112E-17
0.9504 AIR	4.3450E-05	65	0.180556	-3.2E-06	2.98E-06	-6.1E-07	3.97E-08	-9.1E-10	7.86E-12	5.7986E-12	6.3784E-17
	4.3450E-05	70	0.194444	-1.2E-06	1.04E-06	-1.8E-07	9.2E-09	-1.6E-10	9.18E-13	5.8748E-12	6.4623E-17
	4.3450E-05	75	0.208333	-4.7E-07	3.62E-07	-5.3E-08	2.13E-09	-2.6E-11	1.07E-13	5.9113E-12	6.5025E-17
	4.3450E-05	80	0.222222	-1.8E-07	1.26E-07	-1.6E-08	4.92E-10	-4.5E-12	1.25E-14	5.9282E-12	6.5211E-17
	4.3450E-05	85	0.236111	-7E-08	4.4E-08	-4.6E-09	1.14E-10	-7.6E-13	1.46E-15	5.9358E-12	6.5294E-17
	4.3450E-05	90	0.25	-2.7E-08	1.53E-08	-1.3E-09	2.64E-11	-1.3E-13	1.71E-16	5.9392E-12	6.5331E-17
	4.3450E-05	95	0.263889	-1.1E-08	5.35E-09	-4E-10	6.1E-12	-2.2E-14	1.99E-17	5.9406E-12	6.5347E-17
	4.3450E-05	100	0.277778	-4.1E-09	1.87E-09	-1.2E-10	1.41E-12	-3.8E-15	2.33E-18	5.9412E-12	6.5354E-17
	4.3450E-05	150	0.416667	-3E-13	4.97E-14	-5.6E-16	6.21E-19	-7.6E-23	1.1E-27	5.9417E-12	6.5358E-17
	4.3450E-05	200	0.555556	-2.2E-17	1.32E-18	-2.7E-21	2.73E-25	-1.5E-30	5.2E-37	5.9417E-12	6.5358E-17
	4.3450E-05	250	0.694444	-1.7E-21	3.52E-23	-1.3E-26	1.2E-31	-3.1E-38	2.46E-46	5.9417E-12	6.5358E-17
	4.3450E-05	300	0.833333	-1.2E-25	9.37E-28	-6.3E-32	5.3E-38	-6.4E-46	1.16E-55	5.9417E-12	6.5358E-17
	4.3450E-05	350	0.972222	-9.1E-30	2.49E-32	-3E-37	2.33E-44	-1.3E-53	5.48E-65	5.9417E-12	6.5358E-17
	4.3450E-05	400	1.111111	-6.8E-34	6.63E-37	-1.5E-42	1.03E-50	-2.6E-61	2.59E-74	5.9417E-12	6.5358E-17
	4.3450E-05	450	1.25	-5E-38	1.77E-41	-7E-48	4.52E-57	-5.3E-69	1.22E-83	5.9417E-12	6.5358E-17
	4.3450E-05	500	1.388889	-3.7E-42	4.7E-46	-3.4E-53	1.99E-63	-1.1E-76	5.78E-93	5.9417E-12	6.5358E-17
	4.3450E-05	550	1.527778	-2.8E-46	1.25E-50	-1.6E-58	8.75E-70	-2.2E-84	2.7E-102	5.9417E-12	6.5358E-17
	4.3450E-05	600	1.666667	-2.1E-50	3.33E-55	-7.8E-64	3.85E-76	-4.5E-92	1.3E-111	5.9417E-12	6.5358E-17
	4.3450E-05	650	1.805556	-1.5E-54	8.86E-60	-3.8E-69	1.69E-82	-9E-100	6.1E-121	5.9417E-12	6.5358E-17
	4.3450E-05	700	1.944444	-1.1E-58	2.36E-64	-1.8E-74	7.46E-89	-2E-107	2.9E-130	5.9417E-12	6.5358E-17
	4.3450E-05	1000	2.777778	-1.9E-83	8.38E-92	-2E-106	5.4E-127	-1E-153	3.2E-186	5.9417E-12	6.5358E-17
	4.3450E-05	1300	3.611111	-3E-108	3E-119	-3E-138	3.9E-165	-9E-200	3.6E-242	5.9417E-12	6.5358E-17
	4.3450E-05	1600	4.444444	-5E-133	1.1E-146	-4E-170	2.9E-203	-6E-246	4E-298	5.9417E-12	6.5358E-17
	4.3450E-05	1900	5.277778	-9E-158	3.8E-174	-4E-202	2.1E-241	-4E-292	0	5.9417E-12	6.5358E-17
	4.3450E-05	2200	6.111111	-1E-182	1.3E-201	-6E-234	1.5E-279	0	0	5.9417E-12	6.5358E-17
	4.3450E-05	2500	6.944444	-2E-207	4.7E-229	-7E-266	0	0	0	5.9417E-12	6.5358E-17
	4.3450E-05	2800	7.777778	-4E-232	1.7E-256	-9E-298	0	0	0	5.9417E-12	6.5358E-17
	4.3450E-05	3100	8.611111	-7E-257	6E-284	0	0	0	0	5.9417E-12	6.5358E-17
	4.3450E-05	3400	9.444444	-1E-281	0	0	0	0	0	5.9212E-12	6.5133E-17
	4.3300E-05	3700	10.27778	-2E-306	0	0	0	0	0	5.9212E-12	6.5133E-17
	4.191E-05	36500	100	0	0	0	0	0	0	5.7311E-12	6.3042E-17
	2.9700E-06	3650000	10000	0	0	0	0	0	0	4.0614E-13	4.4675E-18

ID	ATWT (G/MOL)	T1/2(YR)	SILO3 CONC (pCi/in³ SOLID)	CONC (Lb/in³ SOLID)	CONC (lb/ft3 SOLID)	MASS/BOREHOLE(LB/FT³) 117.1 bh	Ci waste/borehole
U238	238	4.4700E+09	122958	8.0707E-04	1.3946E+00	2.184E-01	0.1306767838908
TH234	234	6.6000E-02	122958	1.1716E-14	2.0246E-11	3.171E-12	0.1306767838908
PA234	234	7.6400E-04	122958	1.3562E-16	2.3436E-13	3.670E-14	0.1306767838908
U234	234	2.4600E+05	78787	2.7982E-08	4.8353E-05	7.573E-06	0.08373291508
TH230	230	7.5400E+04	6466625	6.9191E-07	1.1956E-03	1.873E-04	6.8725724038119
RA226	226	1.6000E+03	214037	4.7752E-10	8.2515E-07	1.292E-07	0.2274733388119
RN222	222	1.0000E-02	214037	2.9317E-15	5.0659E-12	7.934E-13	0.2274733388119
RB210	210	2.2300E+01	446283	1.2895E-11	2.2282E-08	3.490E-09	0.4742987617328
U235	235	7.0400E+08	5327	5.4374E-06	9.3959E-03	1.472E-03	0.0056614065599
TH231	231	3.0000E-03	5327	2.2776E-17	3.9358E-14	6.164E-15	0.0056614065599
PA231	231	3.2800E+04	45531	2.1285E-09	3.6780E-06	5.760E-07	0.0483892438665
AC227	227	2.1770E+01	42122	1.2843E-12	2.2193E-09	3.476E-10	0.0447662412454
TH227	227	5.1000E-02	41532	2.9665E-15	5.1262E-12	8.028E-13	0.0441392035374
RA223	223	3.1300E-02	31879	1.3729E-15	2.3723E-12	3.715E-13	0.0338802289697
TH232	232	1.4000E+10	38336	7.6823E-04	1.3275E+00	2.079E-01	0.0407425721567
RA228	228	5.7600E+00	18488	1.4980E-13	2.5886E-10	4.054E-11	0.0196485985505
AC228	228	7.0400E-04	18488	1.8309E-17	3.1638E-14	4.955E-15	0.0196485985505
TH228	228	1.9130E+00	34026	9.1565E-14	1.5823E-10	2.478E-11	0.0361620085612
RA224	224	1.0000E-02	18209	2.5165E-16	4.3486E-13	6.811E-14	0.0193520841089
PB212	212	1.2000E-03	18209	2.8581E-17	4.9388E-14	7.735E-15	1.0193520841089
PO210	210	3.7900E-01	4196	2.0605E-15	3.5605E-12	5.576E-13	0.004459407157

Appendix 3: Spreadsheet Output for Analytical Model of 222-Rn Diffusion: K-65 and Silo 3 Wastes

Intentionally left blank

FIXED CONC. LOWER BC AND ZERO CONC. UPPER BC, W/DECAY

PA2 MODEL adapted for Rn222

0.0000142784112129 flux based on 0 conc. in atm. at UBC

K-65	initial conc.	time		n										
	Ci/m^3	d	yr	1	2	3	4	5	6	7	C(x) [Ci/m^3]	FLUX	Da[Ci/M^2 s]	flux [Ci/m^2 s]
dist. to surface	4.3450e-05	30	0.0833333	0	6.68330152E-40	1.009541E-84	-3.3126E-148	-1.9374E-229	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
m	4.3450e-05	35	0.0972222	-5.4440413E-14	2.57029164E-46	1.169456E-98	-1.1336E-172	-1.9207E-267	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
1.219	4.3450E-05	40	0.11111111	-6.8169764E-16	9.88493350E-53	1.35470E-112	-3.8791E-197	-1.9041E-305	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	45	0.125	-8.5361526E-18	3.80158846E-59	1.56929E-126	-1.3274E-221	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
EFFECTIVE D	4.3450E-05	50	0.13888889	-1.0688889E-19	1.46203056E-65	1.81787E-140	-4.5423E-246	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
M^2/D	4.3450E-05	55	0.15277778	-1.3384524E-21	5.62273736E-72	2.10583E-154	-1.5544E-270	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
0.3456 SOIL	4.3450E-05	60	0.16666667	-1.6759973E-23	2.16241550E-78	2.43939E-168	-5.3190E-295	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
0.9504 AIR	4.3450e-05	65	0.18055556	-2.0986678E-25	8.31630666E-85	2.82580E-182	-1.8201E-319	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450e-05	70	0.19444444	-2.6279317E-27	3.19831948E-91	3.27342E-196	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	75	0.20833333	-3.2906709E-29	1.23002289E-97	3.79193E-210	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	80	0.22222222	-4.1205467E-31	4.7304727E-104	4.39259E-224	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	85	0.23611111	-5.1597091E-33	1.8192647E-110	5.08839E-238	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	90	0.25	-6.4609384E-35	6.9966031E-117	5.89440E-252	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	95	0.26388889	-8.0903252E-37	2.6907825E-123	6.82809E-266	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	100	0.27777778	-1.0130628E-38	1.0348322E-129	7.90968E-280	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	150	0.41666667	-9.6015239E-58	7.3245917E-194	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	200	0.55555556	-9.1000541E-77	5.1843810E-258	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	250	0.69444444	-8.6247752E-96	3.6560858E-322	0	0	0	0	0	1.4278e-05		-1.5706e-10	1.5706e-10
	4.3450e-05	300	0.83333333	-8.174319E-115	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	350	0.97222222	-7.747390E-134	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	400	1.11111111	-7.342758E-153	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	450	1.25	-6.959259E-172	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	500	1.38888889	-6.595790E-191	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	550	1.52777778	-6.251304E-210	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	600	1.66666667	-5.924810E-229	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	650	1.80555556	-5.615368E-248	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	700	1.94444444	-5.322088E-267	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	1000	2.77777778	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	1300	3.61111111	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	1600	4.44444444	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	1900	5.27777778	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	2200	6.11111111	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	2500	6.94444444	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	2800	7.77777778	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3450E-05	3100	8.61111111	0	0	0	0	0	0	0	1.4278E-05		-1.5706E-10	1.5706E-10
	4.3300E-05	3400	9.44444444	0	0	0	0	0	0	0	1.4229E-05		-1.5652E-10	1.5652E-10
	4.3300E-05	3700	10.2777778	0	0	0	0	0	0	0	1.4229E-05		-1.5652E-10	1.5652E-10
	0.000042	36500	100	0	0	0	0	0	0	0	1.3772E-05		-1.5150E-10	1.5150E-10
	2.9700E-06	3650000	10000	0	0	0	0	0	0	0	9.7599e-07		-1.0736E-11	1.0736e-11

FIXED CONC. LOWER BC AND ZERO CONC. UPPER BC, W/DECAY PA2 MODEL adapted for Rn222

flux based on 0 conc. in atm. at UBC

Silo 3	initial conc.	time	n									
	Ci/m³	d	yr	1	2	3	4	5	6	7	C(x)	flux [Ci/m² s]
dist to surface	4.3000E-07	30	0.083333	-0.00245	0.004741	-0.00324	0.001119	-0.00022	2.7E-05	-2E-06	1.1E-14	1.2E-19
m	4.3000E-07	35	0.097222	-0.00095	0.001654	-0.00095	0.000259	-3.8E-05	3.1E-06	-1.5E-07	1.8E-14	2E-19
21.336	4.3000E-07	40	0.111111	-0.00037	0.000577	-0.00028	6E-05	-6.4E-06	3.6E-07	-1.1E-08	3E-14	3.3E-19
	4.3000E-07	45	0.125	-0.00014	0.000201	-8.2E-05	1.4E-05	-1.1E-06	4.2E-08	-8.3E-10	4E-14	4.4E-19
effective D	4.3000E-07	50	0.138889	-5.5E-05	7E-05	-2.4E-05	3.2E-06	-1.9E-07	4.9E-09	-6.2E-11	4.8E-14	5.3E-19
m²/d	4.3000E-07	55	0.152778	-2.1E-05	2.4E-05	-7.1E-06	7.4E-07	-3.2E-08	5.8E-10	-4.7E-12	5.3E-14	5.8E-19
0.3453 SOIL	4.3000E-07	60	0.166667	-8.2E-06	8.5E-06	-2.1E-06	1.7E-07	-5.4E-09	6.7E-11	-3.5E-13	5.6E-14	6.1E-19
0.9504 AIR	4.3000E-07	65	0.180556	-3.2E-06	3E-06	-6.1E-07	4E-08	-9.1E-10	7.9E-12	-2.6E-14	5.7E-14	6.3E-19
	4.3000E-07	70	0.194444	-1.2E-06	1E-06	-1.8E-07	9.2E-09	-1.6E-10	9.2E-13	-2E-15	5.8E-14	6.4E-19
	4.3000E-07	75	0.208333	-4.7E-07	3.6E-07	-5.3E-08	2.1E-09	-2.6E-11	1.1E-13	-1.5E-16	5.9E-14	6.4E-19
	4.3000E-07	80	0.222222	-1.8E-07	1.3E-07	-1.6E-08	4.9E-10	-4.5E-12	1.3E-14	-1.1E-17	5.9E-14	6.5E-19
	4.3000E-07	85	0.236111	-7E-08	4.4E-08	-4.6E-09	1.1E-10	-7.6E-13	1.5E-15	-8.2E-19	5.9E-14	6.5E-19
	4.3000E-07	90	0.25	-2.7E-08	1.5E-08	-1.3E-09	2.6E-11	-1.3E-13	1.7E-16	-6.2E-20	5.9E-14	6.5E-19
	4.3000E-07	95	0.263889	-1.1E-08	5.4E-09	-4E-10	6.1E-12	-2.2E-14	2E-17	-4.6E-21	5.9E-14	6.5E-19
	4.3000E-07	100	0.277778	-4.1E-09	1.9E-09	-1.2E-10	1.4E-12	-3.8E-15	2.3E-18	-3.5E-22	5.9E-14	6.5E-19
	4.3000E-07	150	0.416667	-3E-13	5E-14	-5.6E-16	6.2E-19	-7.6E-23	1.1E-27	-1.9E-33	5.9E-14	6.5E-19
	4.3000E-07	200	0.555556	-2.2E-17	1.3E-18	-2.7E-21	2.7E-25	-1.5E-30	5.2E-37	-1.1E-44	5.9E-14	6.5E-19
	4.3000E-07	250	0.694444	-1.7E-21	3.5E-23	-1.3E-26	1.2E-31	-3.1E-38	2.5E-46	-5.9E-56	5.9E-14	6.5E-19
	4.3000E-07	300	0.833333	-1.2E-25	9.4E-28	-6.3E-32	5.3E-38	-6.4E-46	1.2E-55	-3.3E-67	5.9E-14	6.5E-19
	4.3000E-07	350	0.972222	-9.1E-30	2.5E-32	-3E-37	2.3E-44	-1.3E-53	5.5E-65	-1.8E-78	5.9E-14	6.5E-19
	4.3000E-07	400	1.111111	-6.8E-34	6.6E-37	-1.5E-42	1E-50	-2.6E-61	2.6E-74	-1E-89	5.9E-14	6.5E-19
	4.3000E-07	450	1.25	-5E-38	1.8E-41	-7E-48	4.5E-57	-5.3E-69	1.2E-83	-6E-101	5.9E-14	6.5E-19
	4.3000E-07	500	1.388889	-3.7E-42	4.7E-46	-3.4E-53	2E-63	-1.1E-76	5.8E-93	-3E-112	5.9E-14	6.5E-19
	4.3000E-07	550	1.527778	-2.8E-46	1.3E-50	-1.6E-58	8.7E-70	-2.2E-84	2.7E-102	-2E-123	5.9E-14	6.5E-19
	4.3000E-07	600	1.666667	-2.1E-50	3.3E-55	-7.8E-64	3.8E-76	-4.5E-92	1.3E-111	-1E-134	5.9E-14	6.5E-19
	4.3000E-07	650	1.805556	-1.5E-54	8.9E-60	-3.8E-69	1.7E-82	-9E-100	6.1E-121	-5E-146	5.9E-14	6.5E-19
	4.3000E-07	700	1.944444	-1.1E-58	2.4E-64	-1.8E-74	7.5E-89	-2E-107	2.9E-130	-3E-157	5.9E-14	6.5E-19
	4.3000E-07	1000	2.777778	-1.9E-83	8.4E-92	-2E-106	5.4E-127	-1E-153	3.2E-186	-9E-225	5.9E-14	6.5E-19
	4.3000E-07	1300	3.611111	-3E-108	3E-119	-3E-138	3.9E-165	-9E-200	3.6E-242	-3E-292	5.9E-14	6.5E-19
	4.3000E-07	1600	4.444444	-5E-133	1.1E-146	-4E-170	2.9E-203	-6E-246	4E-298	0	5.9E-14	6.5E-19
	4.3000E-07	1900	5.277778	-9E-158	3.8E-174	-4E-202	2.1E-241	-4E-292	0	0	5.9E-14	6.5E-19
	4.3000E-07	2200	6.111111	-1E-182	1.3E-201	-6E-234	1.5E-279	0	0	0	5.9E-14	6.5E-19
	4.3000E-07	2500	6.944444	-2E-207	4.7E-229	-7E-266	0	0	0	0	5.9E-14	6.5E-19
	4.3000E-07	2800	7.777778	-4E-232	1.7E-256	-9E-298	0	0	0	0	5.9E-14	6.5E-19
	4.3000E-07	3100	8.611111	-7E-257	6E-284	0	0	0	0	0	5.9E-14	6.5E-19
	4.8000E-07	3400	9.444444	-1E-281	0	0	0	0	0	0	6.6E-14	6.5E-19
	4.8000E-07	3700	10.27778	-2E-306	0	0	0	0	0	0	6.6E-14	7.2E-19
	9.6E-07	36500	100	0	0	0	0	0	0	0	1.3E-13	1.4E-18
	1.9500E-06	109500	300	0	0	0	0	0	0	0	2.7E-13	2.9E-18
	3.6800E-06	255500	700	0	0	0	0	0	0	0	5E-13	5.5E-18
	4.79E-06	365000	1000	0	0	0	0	0	0	0	6.6E-13	7.2E-18
	6.35E-06	547500	1500	0	0	0	0	0	0	0	8.7E-13	9.6E-18
	7.59E-06	730000	2000	0	0	0	0	0	0	0	1E-12	1.1E-17
	1.191E-05	3650000	10000	0	0	0	0	0	0	0	1.6E-12	1.8E-17
	1.107E-05	7300000	20000	0	0	0	0	0	0	0	1.5E-12	1.7E-17
	1.016E-05	1.1E+07	30000	0	0	0	0	0	0	0	1.4E-12	1.5E-17
	8.54E-06	1.8E+07	50000	0	0	0	0	0	0	0	1.2E-12	1.3E-17
	7.18E-06	2.6E+07	70000	0	0	0	0	0	0	0	9.8E-13	1.1E-17
	5.54E-06	3.7E+07	100000	0	0	0	0	0	0	0	7.6E-13	8.3E-18

Appendix 4: Input Files for the SWIFTII Simulations: K-65 and Silo 3 Wastes

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ONE-DIMENSIONAL DIFFUSION ONLY - 168 CANS/BOREHOLE Fernald K65.new solub.
FIVE MEMBER RADIONUCLIDE CHAIN (CHAIN 1) TORTUOSITY =3 ,325.3 boreholes

```
     4     0     0     0     0     0     0     0     1     0
   148     1     1     1    11     3     0     1     0     0     0     0     1     1
     1     0     1     0     0     0
238 U238              1     0     0  4.470E09
234 U234              2     1     0  2.46E05
     1     1.0
230 TH230             3     1     0  7.54E04
     2     1.0
226 RA226             4     1     0  1600.0
     3     1.0
210 PB210             5     1     0  22.3
     4     1.0
235 U235              6     0     0  7.04E08
231 PA231             7     1     0  3.28E04
     6     1.0
227 AC227             8     1     0  21.77
     7     1.0
232 TH232             9     0     0  1.4E10
228 RA228            10     1     0  5.76
     9     1.0
228 TH228            11     1     0  1.913
    10     1.0
       0.00       0.00       0.00       0.00       0.00       0.00       0.00
       0.00       0.00       0.00       0.00
 0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02
 0.320E-02  0.320E-02  0.320E-02  0.320E-02
 0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02
 0.320E-02  0.320E-02  0.320E-02  0.320E-02
       1.00       1.00       1.00       1.00       1.00       1.00       1.00
       1.00       1.00       1.00       1.00
       1.00       1.00       1.00       1.00       1.00       1.00       1.00
       1.00       1.00       1.00       1.00
       1.00       1.00       1.00       1.00       1.00       1.00       1.00
       1.00       1.00       1.00       1.00
 0.749E-08  0.749E-08  0.749E-08  0.749E-08  0.749E-08  0.749E-08  0.749E-08
 0.749E-08  0.749E-08  0.749E-08  0.749E-08
 0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04
 0.294E-04  0.294E-04  0.294E-04  0.294E-04
 0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04
 0.294E-04  0.294E-04  0.294E-04  0.294E-04
        0.0        0.0        0.0        1.0        1.0        0.0
        1.0        1.0        1.0        0.0        0.0        0.0        0.0
        1.0        1.0        1.0        0.0
        0.0        0.0        0.0        0.0
        1.0        1.0        1.0        0.0
        0.0        0.0        0.0        0.0
        1.0        1.0        1.0        0.0
        0.0        0.0        0.0        0.0
      100.0       14.7       60.0      62.34      62.34
    0     0     0     2
       60.0        1.0       60.0        1.0
        0.0       60.0
       10.0       60.0
    0     0
       60.0       14.7        0.0        0.0
 0.10 0.10 0.20 0.30 0.40 136*0.50 0.40 0.30 0.20 0.10 0.05 0.025 0.025
  1.0
  1.0
   0.1E-13    0.1E-13    0.1E-13    0.0949     0.00000    0.00000    0.00000
    1     1     1     1     1     1
        1.0        1.0        1.0        1.0        0.0        0.0 3.927E+04
        0.0        0.0        0.0
    0     0     0     0     0     0
    0     0
    0     0     0
    1     1     1     1     1     1     1
    2   147     1     1     1     1     2
  148   148     1     1     1     1     3
    0
       0.0
    0
```

```
      0.0        0.0        1.0        0.0
      1.0        1.0        1.0        1.0
   1    1    1    1    1    1
      0.
U238       9.079E-02
U234       5.089E-06
TH230      1.007E-04
RA226      1.314E-05
PB210      7.209E-08
U235       1.186E-03
PA231      2.333E-06
AC227      2.889E-09
TH232      2.748E-01
RA228      1.061E-10
TH228      2.446E-10
0.130E-03 0.740E-08 0.850E-18 0.230E-08 0.390E-19 0.170E-05 0.550E+01
0.550E+01 0.230E-14 0.180E-13 0.210E-23
   1    1    1
 115    1    1
 124    1    1
 130    1    1
 143    1    1
 147    1    1
   0
   0    0    0    0    1    0    0    0    0    0
 1.0E-12    0.0E+00
      10          1        0.0        0.0        0.0        0.0        0.0     0.0
   1   -1   -1   -1   -1   -1  000     0 0000     0   00     0     0     0
  -1 0000 0000 0000 1111    1
   1  146    1    1    1    1
   1  146    1    1    1    1
   1  146    1    1    1    1
   1  146    1    1    1    1
   0    0    0    0    1    0    0    0    0    0
  1.0E-12   0.0E+00
     105          5        0.0        0.0        0.0        0.0        0.0     0.0
   1   -1   -1   -1   -1   -1  000     0 0000     0   00     0     0     0
  -1 0000 0000 0000 1111    0
   0    0    0    0    1    0    0    0    0    0
  1.0E-12   0.0E+00
     365         20        0.0        0.0        0.0        0.0        0.0     0.0
   1   -1   -1   -1   -1   -1  000     0 0000     0   00     0     0     0
  -1 0000 0000 0000 1111    0
   0    0    0    0    1    0    0    0    0    0
  1.0E-12   0.0E+00
    3650        365        0.0        0.0        0.0        0.0        0.0     0.0
   1   -1   -1   -1   -1   -1  000     0 0000     0   00     0     0     0
  -1 0000 0000 0000 1111    0
   0    0    0    0    1    0    0    0    0    0
  1.0E-12   0.0E+00
  2555000       3650        0.0        0.0        0.0        0.0        0.0     0.0
   1   -1   -1   -1   -1   -1  000     0 0000     0   00     0     0     0
  -1 0000 0000 0000 1111    0
   0    0    0    0    1    0    0    0    0    0
  1.0E-12   0.0E+00
365000000      36500        0.0        0.0        0.0        0.0        0.0     0.0
   1   -1   -1   -1   -1   -1  000     0 0000     0   00     0     0     0
  -1 0000 0000 0000 1111    0
   0    0    0    1
```

ONE-DIMENSIONAL DIFFUSION ONLY - 168 CANS/BOREHOLE Fernald SILO3, new sol.
FIVE MEMBER RADIONUCLIDE CHAIN (CHAIN 1) TORTUOSITY =3 , LONG T

```
    4     0     0     0     0     0     0     0     1     0
  148     1     1     1    11     3     0     1     0     0     0     0     1     1
    1     0     1     0     0     0
238 U238              1     0     0  4.470E09
234 U234              2     1     0  2.46E05
    1     1.0
230 TH230             3     1     0  7.54E04
    2     1.0
226 RA226             4     1     0  1600.0
    3     1.0
210 PB210             5     1     0  22.3
    4     1.0
235 U235              6     0     0  7.04E08
231 PA231             7     1     0  3.28E04
    6     1.0
227 AC227             8     1     0  21.77
    7     1.0
232 TH232             9     0     0  1.4E10
228 RA228            10     1     0  5.76
    9     1.0
228 TH228            11     1     0  1.913
   10     1.0
      0.00       0.00       0.00       0.00       0.00       0.00       0.00
      0.00       0.00       0.00       0.00
 0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02
 0.320E-02  0.320E-02  0.320E-02  0.320E-02
 0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02  0.320E-02
 0.320E-02  0.320E-02  0.320E-02  0.320E-02
      1.00       1.00       1.00       1.00       1.00       1.00       1.00
      1.00       1.00       1.00       1.00
      1.00       1.00       1.00       1.00       1.00       1.00       1.00
      1.00       1.00       1.00       1.00
      1.00       1.00       1.00       1.00       1.00       1.00       1.00
      1.00       1.00       1.00       1.00
 0.749E-08  0.749E-08  0.749E-08  0.749E-08  0.749E-08  0.749E-08  0.749E-08
 0.749E-08  0.749E-08  0.749E-08  0.749E-08
 0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04
 0.294E-04  0.294E-04  0.294E-04  0.294E-04
 0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04  0.294E-04
 0.294E-04  0.294E-04  0.294E-04  0.294E-04
       0.0        0.0        0.0        1.0        1.0        0.0
       1.0        1.0        1.0        0.0        0.0        0.0        0.0
       1.0        1.0        1.0        0.0
       0.0        0.0        0.0        0.0
       1.0        1.0        1.0        0.0
       0.0        0.0        0.0        0.0
       1.0        1.0        1.0        0.0
       0.0        0.0        0.0        0.0
     100.0       14.7       60.0      62.34      62.34
   0    0    0    2
      60.0        1.0       60.0        1.0
       0.0       60.0
      10.0       60.0
   0    0
      60.0       14.7        0.0        0.0
 0.10 0.10 0.20 0.30 0.40 136*0.50 0.40 0.30 0.20 0.10 0.05 0.025 0.025
  1.0
  1.0
   0.1E-13    0.1E-13    0.1E-13    0.0949     0.00000    0.00000    0.00000
   1    1     1    1     1    1
      1.0        1.0        1.0        1.0        0.0        0.0 3.927E+04
      0.0        0.0        0.0
   0    0    0    0    0    0
   0    0
   0    0    0
   1    1    1    1    1    1    1
   2  147    1    1    1    1    2
 148  148    1    1    1    1    3
   0
      0.0
   0
```

```
      0.0        0.0        1.0        0.0
      1.0        1.0        1.0        1.0
   1    1    1    1    1    1
      0.
U238       2.184E-01
U234       7.573E-06
TH230      1.873E-04
RA226      1.292E-07
PB210      3.490E-09
U235       1.472E-03
PA231      5.760E-07
AC227      3.476E-10
TH232      2.079E-01
RA228      4.054E-11
TH228      2.478E-11
0.750E-06 0.260E-10 0.210E-09 0.710E-14 0.280E-19 0.500E-08 0.550E+01
0.550E+01 0.230E-06 0.220E-17 0.280E-16
   1    1    1
 118    1    1
 124    1    1
 130    1    1
 143    1    1
 147    1    1
   0
   0    0    0    0    1    0    0    0    0    0
 1.0E-12    0.0E+00
365000000      36500        0.0        0.0        0.0        0.0        0.0    0.0
   1   -1   -1   -1   -1   -1  000     0 0000     0   00     0    0     0
  -1 0000 0000 0000 1111    1
   1  148    1    1    1    1
   1  148    1    1    1    1
   1  148    1    1    1    1
   1  148    1    1    1    1
   0    0    0    1
```

ATTACHMENTS

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Attachment A

A CASE FOR GREATER CONFINEMENT DISPOSAL OF SPECIAL-CASE WASTES AT THE NEVADA TEST SITE

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Sandia National Laboratories

Albuquerque, New Mexico 87185

September 15, 1993

Warren Black
U.S. Department of Energy
EM-322
Trev-II Building
1000 Independence Avenue SW
Washington, DC 20585

Dear Mr. Black:

Subject: Draft of White Paper on GCD Borehole Disposal (Letter/93/S10.1)

Enclosed for your review and comment is a copy of a draft document explaining the advantages of the greater confinement disposal concept for disposal of special-case wastes. This is in response to your request relayed to us by our DOE/NV Technical Program Officer, Joe Ginanni (fax dated 11/9/92), and your direct request at our meeting on September 9, 1993. Please refer to the letter from N. Olague, SNL to you dated February 22, 1993 regarding the original submittal of this document. Please call me at (505)848-0761 or Paul Davis at (505)848-0754 if you have any questions.

Sincerely,

David P. Gallegos
GCD Project Leader
Safety and Risk Assessment
Department 6331

DPG:6331:dpg

Enclosure: as stated

Copy to (w/o enclosure):
6331 P.A. Davis
6331 Day File

Copy to (w/ enclosure):
GCD Records Center (Letter/93/S10.1)
6331 D.P. Gallegos

A CASE FOR GREATER CONFINEMENT DISPOSAL OF SPECIAL-CASE WASTES AT THE NEVADA TEST SITE

Issue

According to the Nuclear Waste Policy Act [1982] as amended, the United States Department of Energy (DOE) is responsible for disposing of high-level radioactive waste, spent nuclear fuel, and radioactive wastes generated as a result of defense-related activities. The total volume and activity for all these wastes is very large.

Radioactive wastes can be categorized as being either spent fuel (fuel rods from a nuclear reactor), high-level waste (liquid or solid waste resulting from reprocessing spent fuel), transuranic waste (alpha-emitting radionuclides with half-lives greater than 20 years and with an atomic number greater than 92 in concentrations greater than 100 nanocuries/gram), or low-level waste (waste that is not spent fuel, high-level waste, or transuranic waste). Regulations for the disposal of spent fuel, high-level waste, and transuranic waste have been promulgated by the Environmental Protection Agency (EPA) in 40 CFR 191 [EPA, 1985]. The Nuclear Regulatory Commission (NRC) regulates the disposal of spent fuel, high-level waste, and *non-defense* transuranic waste in 10 CFR 60 [NRC, 1988a]. The NRC also regulates the disposal of non-defense low-level waste in 10 CFR 61 [NRC, 1988b]. The DOE gives requirements for disposal of all four types of waste in DOE Order 5820.2A [DOE, 1988]. The current disposal strategy is to dispose of all spent fuel and high-level waste (whether defense generated or not) in a high-level radioactive waste repository, to dispose of defense transuranic waste at the Waste Isolation Pilot Plant (WIPP) in southeastern New Mexico, and to dispose of defense low-level waste at various DOE sites around the U.S.

Yucca Mountain, located at the western edge of the Nevada Test Site (NTS), is currently being investigated as a potential high-level radioactive waste repository. This mountain has a very thick unsaturated layer of fractured tuff in which the proposed repository would be located. Proposed designs of the repository consist of mined cavities into which waste is placed, with shafts providing access between the surface and the cavities. This site is expected to provide an effective barrier to radionuclide migration because of the arid environment and the scarcity of water in the rock. However, the physics of flow and transport in unsaturated, fractured, porous rock are not well understood, so it may be difficult (and very expensive) to demonstrate compliance with 40 CFR 191 [EPA, 1985] and 10 CFR 60 [NRC, 1988a] to the satisfaction of the NRC.

The WIPP is located in thick salt beds (610 m) near Carlsbad, New Mexico that are millions of years old. It is designed to hold about 180,000 m^3 of transuranic waste with an activity of about 14 million curies and has cost about $800 million to construct. Most of the rooms into which the defense transuranic waste will be placed have already been mined, although no waste has been emplaced yet. Salt was chosen as a setting for disposal of radioactive waste because it contains very little water and will "creep" in response to the change in

stresses resulting from mining, thereby sealing the waste in place after a few decades. Preliminary calculations [Marietta et al., 1989] have indicated that ground water flow and radionuclide transport through the salt are indeed very slow, making the salt an effective barrier to radionuclide migration. However, the WIPP is located directly under sources of potash and oil, so the potential for human intrusion in the future cannot be excluded. The probability and consequences of human intrusion must be included in the performance assessment that is conducted in accordance with 40 CFR 191. The EPA will certify that the WIPP meets the requirements of 40 CFR 191, as required by the recent WIPP Land Withdrawal legislation.

Most low-level wastes are disposed of using near-surface techniques of disposal, defined by the NRC in 10 CFR 61.2 as disposal in or within the upper 30 meters of the earth's surface (e.g., above-ground vaults, shallow land burial). These disposal techniques are acceptable for low-level waste classes A, B, and C as given by the NRC in 10 CFR 61.55. These classes of waste are defined by their concentrations of certain long-lived and short-lived radionuclides, with class A being the least concentrated and class C being the most concentrated. If the concentration of radionuclides in a given waste is greater than that defined as class C, the waste is known as greater-than-class-C (GTCC) and is not suitable for near-surface disposal.[1] DOE Order 5820.2A states that GTCC waste must be disposed of using alternative disposal methods and that such disposal requires special review and approval. Furthermore, the DOE is responsible for disposing of all GTCC waste, regardless of who generated it.

Much of the radioactive waste in the U.S. is also contaminated with hazardous constituents, as defined by the Resource Conservation and Recovery Act (RCRA). Waste that is both radioactive and hazardous is known as mixed waste, and often poses difficult disposal problems. The hazardous constituents are regulated by the RCRA, and many are prohibited from land disposal unless they are treated.

The disposal strategy outlined above provides disposal options for high-level radioactive waste, spent fuel, defense transuranic waste, or defense low-level waste suitable for near-surface disposal that can meet the waste acceptance criteria for the disposal site for which it is destined. However, a significant portion of the DOE's waste falls outside the scope of this disposal strategy and, thus, has no "official" disposal site. These wastes are known as special-case wastes and include 1) defense transuranic waste that cannot be sent to the WIPP because it does not meet the waste acceptance criteria either for the WIPP or for the TRUPACT-II transport container, 2) non-defense transuranic waste that cannot go to the WIPP because the WIPP will accept only defense transuranic waste, 3) greater-than-class-C

---

[1] Although defense wastes are not subject to 10 CFR 61, in DOE Order 5820.2A the DOE adopts the definition of GTCC that is in 10 CFR 61.55. Accordingly, it will be assumed that the DOE defines near-surface disposal the same way that the NRC does, although we know of no statement to that effect.

low-level wastes, 4) low-level radioactive wastes with radionuclide concentrations that exceed site-specific performance assessment limits for disposal but are not GTCC low-level wastes, and 5) fuel and fuel debris owned by the DOE that may not meet the waste acceptance criteria for a high-level radioactive waste repository.

These special-case wastes come from a variety of sources. Activities that generate special-case wastes include hot cell and fuel fabrication facility decontamination and decommissioning, environmental restoration at various DOE sites, destructive examination of fuels, clean-up of nuclear weapons accidents and the Three Mile Island-2 reactor, isotope separation research, research and development involving nuclear fuel, and retirement of sealed sources and radioisotope thermoelectric generators. Special-case wastes come in many forms, including large metallic structures, reactor vessels, ion-exchange resins, air filters, sludge, gauges, dials, tritiated liquid, six-foot diameter metal spheres, uranium hexafluoride gas cylinders, uranium fuel elements, uranium slugs, and fission chambers. In addition, many of these wastes are contaminated with oils, other organic compounds, and other hazardous materials, making them mixed wastes and subject to various provisions of the Resource Conservation and Recovery Act (RCRA) [Kudera et al., 1990].

Kudera et al. [1990] have estimated the volume of special-case waste to be about 1,080,000 m^3, 80% of which is uncharacterized waste from the tanks at Hanford. This is not the final disposal volume, however, because of anticipated treatment and packaging of waste prior to disposal. Furthermore, this volume estimate does not include the waste expected to be generated by environmental restoration activities, which may produce millions of cubic feet of waste.

The problem of disposal of special-case waste by the DOE is not a small one, and the DOE is in the process of developing strategies to deal with parts of the problem. One of the waste disposal options that should be considered in developing these strategies is called greater confinement disposal (GCD). While GCD cannot be used to dispose of all the special-case waste, it can be used to dispose of a significant fraction of it. The rest of this paper is devoted to justifying the disposal of special-case waste using GCD technology.

Solution

Greater confinement disposal, as practiced at the NTS, consists of 120-foot deep, 10-foot diameter boreholes placed in the alluvium in Frenchman Flat. Waste is placed in the bottom 50 feet and the upper 70 feet is backfilled with alluvium. There are no caps, sleeves, liners, or engineered barriers (a few boreholes have probertite in them to prevent criticality). This method of disposal was developed in the early 1980's in response to the recognition that certain low-level radioactive wastes were not suitable candidates for shallow land disposal. These wastes included greater-than-class-C low-level wastes (discussed above), as well as ^{3}H (tritium, which has a high specific activity and a high environmental mobility), heat-generating wastes, and wastes requiring shielding during disposal operations. To investigate this method of disposal, approximately 1.11 million curies of ^{3}H, ^{60}Co, ^{90}Sr, ^{137}Cs, and ^{226}Ra were placed in the greater confinement disposal test (GCDT) borehole in

1984. This borehole was designed and constructed so that temperature and moisture content could be monitored, and soil gases could be sampled for tritium and for tracers that were intentionally introduced. The conclusion of this work was that GCD boreholes were an effective method of disposing of wastes that were not suitable for shallow land disposal [Dickman, 1989].

Twelve additional GCD boreholes have been drilled since the early 1980's. Four of these holes have transuranic wastes in them, four contain low-level radioactive wastes that are not suitable for near-surface disposal, and four are empty. Chu and Bernard [1991] report the radionuclide inventory in all 13 GCD boreholes at the NTS.

Greater confinement disposal using boreholes is also practiced at the DOE's Savannah River Plant (SRP) site, although the boreholes at SRP are much shallower than those at the NTS because of higher water tables in South Carolina. The SRP boreholes are 34 feet deep and have a 4-foot deep layer of gravel on the bottom. The waste is placed in 20-foot long, 7-foot diameter fiberglass liners that have been grouted in place above the gravel. Once the liner is filled with waste, the borehole is backfilled with layers of concrete, clay, and soil.

Boreholes for waste disposal have been investigated by the Atomic Energy of Canada [Fereday, 1982] and the Dutch [Onshore Disposal Committee, 1989]. Experience with this method has been positive.

There are other disposal technologies besides boreholes that provide greater confinement than shallow-land burial. These technologies include 1) earth covered tumuli, 2) concrete structures, both above and below grade, 3) deep trenches, 4) rock cavities, 5) abandoned mines, 6) high-integrity containers, and 7) hydrofracture [Trevorrow and Schubert, 1989]. Only two of these technologies are likely to be acceptable methods for disposing of special-case wastes: deep trenches and abandoned mines. A third technology, rock cavities (defined by Trevorrow and Schubert as cavities mined from rock at depths greater than 165 feet specifically for disposal of waste) would probably be acceptable, but this technology sounds very similar to that used at the WIPP and the proposed design at Yucca Mountain. While rock cavities would provide much greater confinement than shallow-land burial, such an expensive means of disposal is probably not needed for most of the special-case waste and it would be redundant to consider it in studying disposal options for special-case waste. Reasons for rejecting the other four methods are discussed below.

If we accept the NRC's definition of near-surface disposal (i.e., in or within the upper 30 m of the earth's surface), then earth-covered tumuli and concrete structures would not be acceptable for disposal of special-case wastes, which are mostly GTCC low-level wastes and transuranic wastes. In fact, these disposal methods are gaining popularity for disposal of low-level waste that *is* acceptable for near-surface disposal, so GCD techniques would probably need to offer more isolation than these two disposal methods.

High-integrity containers are designed to dispose of special-case waste using near-surface disposal techniques; the greater confinement comes from the container, not the disposal method. These containers probably would not be acceptable for disposing of special-case waste because of the restriction on near-surface disposal of GTCC low-level waste.

Finally, hydrofracture consists of injecting a slurry of waste and cement into rock that has horizontal fractures (either natural or induced by high-pressure injection of water). The slurry will solidify as a thin sheet in the rock. This method of disposal would almost certainly not be an acceptable method of disposal of any type of radioactive waste because it would probably constitute a Class IV injection well, which is prohibited (40 CFR 144.13).

Thus, there are at least three disposal technologies that would provide greater confinement disposal of special-case wastes: boreholes, deep trenches, and existing abandoned mines. Our experience has been with the boreholes at the NTS; therefore, our discussion will assume that GCD technology is practiced at an arid site, such as Frenchman Flat at the NTS, using boreholes. However, many of the arguments in favor of GCD using boreholes apply to deep trenches and abandoned mines also.

The primary reason that GCD boreholes are a good method for disposing of special-case waste is that it is safe: we believe that it can meet pre- and post-closure safety requirements. For example, Dickman [1989] reported that the waste handling systems used for the GCDT afforded excellent radiological protection. Borehole disposal is compatible with remote waste emplacement techniques, and in over 100 remote-handling GCD operations conducted at the NTS, none resulted in recordable exposures to personnel. Dickman also reported that doses resulting from GCD boreholes were less than those resulting from shallow land burial, as calculated by a long-term risk assessment. This high level of worker safety results, in part, from the shielding provided by the sides of the borehole and the fact that personnel do not enter the borehole, as they would a trench or cavity. Because the operating time is short for individual holes, closing individual holes does not affect nearby holes, and the small cap area simplifies final closure. In addition, it is relatively easy to temporarily shelter the open hole from weather during the operational phase of the facility. A preliminary performance assessment concluded that greater confinement disposal of transuranic waste in boreholes at the NTS was likely to meet the requirements of 40 CFR 191 [Price et al., 1992], the primary post-closure safety requirement for transuranic waste.

It makes sense that greater confinement disposal would isolate wastes from the public more effectively than shallow land burial, simply because the depth of burial is greater with the former than with the latter. Also, the small area at the land surface reduces vulnerability to cap failure compared to a trench design and to surface phenomena such as erosion and flooding. It is not obvious, however, why GCD boreholes could be just as safe during the post-closure period as a deep geologic repository, such as the WIPP or the proposed repository at Yucca Mountain. The answer lies in the site itself: there are very few processes that can transport waste to the accessible environment or to a member of the public. First, the alluvium in Frenchman Flat is very dry and the little precipitation that

falls in the region evaporates before it can reach any significant depth, so convective transport of waste is expected to be negligible or nonexistent. Diffusion is the only other process by which radionuclides can migrate, and this is a very slow process in the liquid phase. Diffusion is somewhat faster for vapor phase radionuclides (e.g., ^{3}H, ^{14}C, ^{222}Rn), but not so fast as to be a problem. Second, plant populations are very sparse and are not of the variety eaten by humans, so radionuclides are not likely to reach humans via the food chain. Third, any tectonic activity in the region would probably have minimal consequences because alluvium cannot support open fractures. Fourth, the borehole design minimizes the "footprint" of the disposal system (because of the small area at the land surface) so that the probability of inadvertent human intrusion is minimized relative to other disposal configurations, such as mined repositories that have a larger "footprint," or area for intrusion. Finally, the geologic setting is such that potential future changes, natural or man made, would have little effect on the system performance.

Another reason that GCD boreholes are a good method of disposing of special-case wastes is that it is very cost effective. Dickman et al. [1984] estimated the cost of GCD boreholes to range from $600 to $2120 per cubic meter. This cost included site preparation, construction, and waste loading for both contact- and remote-handled wastes. In contrast, the cost of disposal for the WIPP is approximately $4322 per cubic meter (for site preparation and construction only) while the projected cost of disposal at the proposed repository at Yucca Mountain is _____.

A third reason that GCD boreholes are a good method of disposing of special-case wastes is that the site (Frenchman Flat at the NTS) is simple in terms of being able to model it for performance assessment purposes. This may seem to be a trivial point, but it is not. A simple site is not only much less expensive to investigate and characterize, but it also affords a stronger basis for providing "reasonable assurance" that the site will meet the requirements of 40 CFR 191. The alluvium into which the GCD boreholes are drilled is unfractured, porous, unsaturated, and relatively homogeneous, so that it is uncomplicated to develop and implement models for ground-water flow and radionuclide transport. In contrast, a site located in fractured, unsaturated tuff (such as Yucca Mountain) is difficult to model because our understanding of flow and transport in fractured unsaturated rock is limited and is hotly debated within the scientific community. Therefore, even though millions of dollars may be spent studying flow and transport in fractured unsaturated rock, it may be difficult to demonstrate compliance with 40 CFR 191 and 10 CFR 60 to the NRC's satisfaction because the site itself is so complex and difficult to model.

Another point in favor of disposing of special-case wastes using GCD boreholes is that engineered barriers are not needed, but are very simple to use, if desired; no special grouts or seals are needed. None of the GCD boreholes that contain waste have any engineered barriers (except for three boreholes which were backfilled with probertite to eliminate the possibility of nuclear criticality). In fact, in the unsaturated zone, engineered barriers such as liners could actually hurt the performance of the GCD boreholes by creating a "bathtub" in which water collects. However, the technology itself is amenable to engineered barriers,

such as layering materials of different grain sizes to provide a capillary barrier, or pouring concrete around the waste to provide a pH buffer.

A final reason that GCD boreholes are a good method of disposing of special-case wastes is that it is also a good place to dispose of solid hazardous waste that is land-disposal restricted (LDR) under the RCRA and is typically mixed with radioactive waste. The DOE currently plans to treat all of its LDR waste so that it is no longer restricted from land disposal, but it is likely that, for some LDR wastes, treatment technologies will not exist or will be far too costly to implement. For some of these wastes, GCD boreholes at the NTS would probably be a good disposal method because the characteristics of the site (discussed above) make it a good candidate for a no-migration variance, which must be obtained if untreated LDR waste is disposed of.

The amount of waste that could be disposed of using GCD boreholes is limited by the diameter of the borehole, and many boreholes would be needed for a large volume of waste. There is also a limit on how deep the boreholes can go, both from an operational point of view and from a performance assessment point of view. Thus, some of the DOE's special-case waste can be disposed of using GCD borehole technology, although the exact amount will be depenent on the amount of land available.

In summary, greater confinement disposal should be part of the DOE's strategy for disposing of its special-case wastes. Low-level radioactive wastes that cannot or should not be disposed of in the near surface (i.e., GTCC, high specific activity, thermal sources, and wastes requiring shielding during handling operations) and special-case transuranic wastes are excellent candidates for GCD boreholes because this disposal method can safely, cost-effectively, and legally dispose of these wastes.[2] This conclusion is supported by the results of a preliminary 40 CFR 191 compliance assessment [Price et al., 1992], experience with this type of disposal system [Dickman, 1989], and the Canadian [Fereday, 1982] and Dutch [Onshore Disposal Committee, 1989] experiences. Sandia National Laboratories is beginning a system configuration study to examine GCD-type disposal methods and waste types that might be good candidates for greater confinement disposal at the NTS.

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[2] Spent fuel, high-level radioactive waste, and transuranic waste do not have to be disposed of in deep geologic repositories. Although the DOE's strategy is to dispose of these types of waste in deep geologic repositories, this type of disposal is not required by 40 CFR 191. It does make sense, however, to dispose of these wastes at depths greater than 30 meters, the NRC's limit for near-surface disposal.

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Attachment B

Memorandum of January 12, 1995 from Carl Gertz (Director, DOE/NV/WMD) to David Gallegos (Sandia National Laboratory, Org. 6331)

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Department of Energy
Nevada Operations Office
P.O. Box 98518
Las Vegas, NV 89193-8518

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David P. Gallegos
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REQUEST FOR EVALUATION OF GREATER CONFINEMENT DISPOSAL OF FERNALD ENVIRONMENTAL MANAGEMENT PROJECT 11(e)2 BYPRODUCT WASTE

Fernald Environmental Management Project (FEMP) has contacted our office to discuss the potential of disposal of vitrified 11(e)2 byproduct material from their K-65 and cold metal oxide silos. This material is part of the FEMP Operable Unit (OU) 4 cleanup activity. Enclosed you will find the following information that provides specific information on OU4 waste.

1. Memorandum, J. P. Hamric to N. C. Aquilina, dated February 11, 1994.
2. Memorandum, J. E. Lytle to J. N. Fiore, dated November 16, 1994.
3. Nevada Test Site (NTS) comments on OU 4 Feasibility Study/Proposed Plan with FEMP responses, dated November 30, 1993.
4. Draft Waste Stream Information Section for the FEMP application according to the Nevada Test Site Defense Waste Acceptance Criteria, Certification, and Transfer Requirements, NVO-325.

Please conduct a review of this information and provide to our office a determination as to whether this waste could be disposed of in the Greater Confinement Disposal (GCD) units and, if necessary, what issues need to be resolved. Where possible provide potential options for identified issues. The results of your evaluation should be submitted to this office by March 1, 1995.

For the purposes of this evaluation, one assumption to use is that any OU4 waste being sent to NTS meets the requirements of NVO-325. Also, consider any effect the disposal of this waste would have on other waste management activities such as monitoring, permitting (underground injection control), and closure of disposal unit. If necessary, coordinate the summary of GCD options and/or issues with personnel from other organizations as appropriate. The regulatory concerns of the state of Nevada will be addressed after the technical issues and options have been identified.

In order to evaluate all possible disposal configuration options, a copy of this information was also forwarded to Reynolds Electrical and Engineering Co., Inc., for their evaluation on disposal configuration options for this waste.

If you have any questions or need additional information to complete your evaluation, please contact Wendy A. Griffin, Waste Management Division, at (702) 295-5751.

Carl P. Gertz, Director
Waste Management Division

WMD:WAG

Enclosures:
As stated

cc w/o encls:
S. P. Cowan, DOE/HQ
(EM-30) TREV
W. D. Black, DOE/HQ
(EM-351) TREV
B. D. Becker, REECo/WOD,
Mercury, NV

DOE F 1320.8 (6-90)

United States Government **Department of Energy**

Fernald Field Office

memorandum

FEB 1 1 1994

DATE: DOE-0846-94

REPLY TO ATTN OF: FN:Allen

SUBJECT: REQUEST FOR APPROVAL TO DISPOSE OF 11(e)2 BYPRODUCT MATERIAL AT THE NEVADA TEST SITE

TO: Nick C. Aquilina, Manager, Nevada Operations Office

SUMMARY

In accordance with DOE Order 5820.2A, "Radioactive Waste Management," the Fernald Environmental Management Project (FEMP) requests your approval for disposal, after treatment, of the 11(e)2 byproduct material contained in Fernald's K-65 and cold metal oxide silos at the Nevada Test Site (NTS). The leading remedial alternative for material treatment is vitrification.

The K-65 and cold metal oxide silos (Silos 1, 2 and 3) contain approximately 14,000 cu yds of uranium ore residues and cold metal oxides (estimated to be less than 6,000 cu yd after vitrification) classified under the Atomic Energy Act as 11(e)2 byproduct material. DOE Order 5820.2A provides for disposal of small volumes of 11(e)2 byproduct material at DOE low-level waste sites with approval of the appropriate field organization. FEMPs interpretation of the intent of the "small volume" requirement in DOE Order 5820.2A was to preclude disposal of the large volume of uranium tailings from throughout the complex at low-level waste disposal facilities. As described below, the 11(e)2 byproduct material in the silos meets the description of a small volume and therefore may be disposed at NTS. The Manager of the Nevada Operations Office is the appropriate approval authority.

We seek your approval on whether this 11(e)2 byproduct material can be disposed at the NTS pursuant to DOE Order 5820.2A, Chapter IV, and, if approved, that Fernald can pursue qualification of the vitrified silo residues as an approved waste stream in the regular manner under NVO-325, the NTS waste acceptance criteria document.

DISPOSAL OF BYPRODUCT MATERIAL AS LLW

The FEMP's silo residues meet the definition of byproduct material in section 11(e)2 of the Atomic Energy Act, in that they are ". . .tailings or waste produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content" (the Attachment to this memorandum provides information specific to the FEMP residues as to their origin and processing prior to placement in the silos). DOE Order 5820.2A, "Radioactive Waste Management," dated September 26, 1988, Chapter III, provides for the disposal of low-level

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radioactive waste (LLW) at DOE sites pursuant to certain requirements, but the definition of LLW specifically excludes 11(e)2 byproduct material. However, Chapter IV of the Order sets forth the policy that "small volumes of DOE waste containing 11(e)2 byproduct material...may be managed as low-level waste in accordance with the requirements of Chapter III of this Order."

As stated above, the reference to "small volumes" is intended to prevent the disposal of large volumes of uranium mill tailings (millions of cubic yards) from consuming the limited resources of LLW disposal facilities at DOE sites. Moreover, Chapter IV also states, "Waste covered under this chapter in quantities too large for acceptance at DOE low-level waste disposal sites shall be...disposed of at specially designated DOE sites or tailing disposal sites established under the Uranium Mill Tailings Radiation Control Act of 1978."

The test for "small volumes", therefore, is defined by the FEMP to be quantities that are not "too large for acceptance at DOE low-level waste disposal sites." The current volume of the FEMP silo residues is approximately 14,000 cu yd. Vitrification is expected to reduce this volume to less than 6,000 cu yd. The latter represents only 15 to 20 percent of the LLW disposed of annually at the NTS in recent years. In addition, exhumation and treatment, and therefore shipment of the residues is currently scheduled to occur over a three year period.

Documentation meeting the requirements of the National Environmental Policy Act for the remediation of the silos, and the vitrification, packaging, and shipment of the silo residues, is contained in our Operable Unit 4 Feasibility Study/Proposed Plan-Environmental Impact Statement.

Your approval of this request will enable us to develop a waste application which will support implementation of the preferred remedial alternative identified in the Proposed Plan for Operable Unit 4, and assist in gaining the required Environmental Protection Agency approval. The Feasibility Study and Proposed Plan are scheduled for final approval by the United States Environmental Protection Agency (Region V) on January 26, 1994. Approval of the Feasibility Study and Proposed Plan will be followed by a public review period beginning March 7, 1994, at which time the public has an opportunity to comment on the preferred remedial alternative. Following the public review period, the Record of Decision will be completed which will address public input on the remediation plans and identify the selected alternative for Operable Unit 4. Therefore, to support the next step in the remediation of Operable Unit 4, we request your completed evaluation by March 4, 1994, before the Public Comment period begins.

While your approval to dispose of these 11(e)2 byproduct materials at the NTS is not alone sufficient authorization to begin shipments, it is a necessary step toward that objective. Prior to shipment of these waste materials to the NTS for disposal, DOE-FN and its operating contractor will, in the regular, prescribed manner, apply for DOE-NV approval of the silo residues as a waste stream under NVO-325 waste acceptance criteria. These criteria include, among other things, prohibitions of certain constituents, limits on surface radiation dose rates, packaging and

certification requirements—all of which are familiar to the FEMP due to our successful program of adherence to NVO-325 in the management of our LLW shipped to Nevada for disposal. Your organization's approval of this waste stream under NVO-325 will be required before shipments can begin.

Once the NVO-325 criteria are successfully met and demonstrated, we will work closely with your office and your operating contractor to determine the appropriate material handling requirements and disposal configuration for these residues.

An attachment is provided for additional information related to the remedial plans for the silo residues. Items in the attachment include background on the history of the silos and silo residue characteristics; description of the Operable Unit 4 preferred remedial alternative and basis for selection; description of the vitrification process; characteristics of the vitrified product for disposal; and the schedule for remediation activities.

This letter was coordinated with Wendy Griffin of your staff, and her comments have been incorporated.

ACTION

We are seeking your approval by March 4, 1994, to dispose of the FEMP 11(e) 2 byproduct material, after treatment, at NTS.

J. Phil Hamric
Manager

Attachment: As Stated

APPROVED:

Nick C. Aquilina, Manager Date
Nevada Operations Office

cc w/att:

K. A. Chaney, EM-424, TREV
S. Fauver, EM-424, TREV
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