**CBU-PIT-2005-00146**
**Revision 0**

# SALTSTONE
# PERFORMANCE OBJECTIVE DEMONSTRATION DOCUMENT
# (U)

**PREPARED BY:**
**Kent H. Rosenberger**
**Bernice C. Rogers**
**R. Kim Cauthen**

**JUNE 2005**



Westinghouse Savannah River Company
Savannah River Site
Aiken, SC 29808

**Prepared for the U.S. Department of Energy Under**
**Contract Number DE-AC09-96SR18500**

**DISCLAIMER**

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## REVIEWS AND APPROVALS

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## LIST OF ACRONYMS

| | |
|---|---|
| ACRI | Analytic and Computational Research, Inc. |
| ALARA | As Low As Reasonably Achievable |
| ARP/MCU | Actinide Removal Process Modular CSSX Unit |
| ASME | American Society of Mechanical Engineering |
| CAP88 | Clean Air Act Assessment Package-1988 |
| CEDE | Committed Effective Dose Equivalent |
| CFR | Code of Federal Regulations |
| CHA | Consolidated Hazards Analysis |
| CSSX | Caustic Side Solvent Extraction |
| DCF | Dose Conversion Factor |
| DDA | Deliquification Dissolution and Adjustment |
| DOE | Department of Energy |
| DSA | Documented Safety Analysis |
| DRF | Dose Release Factors |
| DWPF | Defense Waste Processing Facility |
| EPA | Environmental Protection Agency |
| ETP | Effluent Treatment Process |
| FGR | Federal Guidance Report |
| GCL | Geosynthetic Clay Liner |
| GSA | General Separations Area |
| HELP | Hydrologic Evaluation of Landfill Performance |
| IC | Institutional Control |
| ICRP | International Commission on Radiological Protection |
| IRRIDOSE | Computer code |
| ISMS | Integrated Safety Management System |
| ISWLF | Individual Solid Waste Landfill |
| ITP | In Tank Precipitation |
| LADTAP | Computer code |
| LADTAP XL | Spreadsheet version of computer code |
| MCU | Modular Caustic side solvent extraction Unit |
| MEI | Maximally Exposed Individual |
| MMES | Martin Marietta Energy Systems, Inc. |
| MST | Monosodium Titanate |
| NDAA | National Defense Authorization Act |
| NCRP | National Council on Radiation Protection and Measurements |
| NQA | Nuclear Quality Assurance |
| NRC | Nuclear Regulatory Commission |
| OSHA | Occupational Safety and Health Administration |
| PA | Performance Assessment |
| PEL | Permissible Exposure Limit |

| | |
|---|---|
| PODD | Performance Objective Demonstration Document |
| PORFLOW | Computer code |
| QA | Quality Assurance |
| QA/QC | Quality Assurance/Quality Control |
| QAMP | Quality Assurance Management Plan |
| RPP | Radiation Protection Porgram |
| SA | Special Analysis |
| SAIC | Science Applications International Corporation |
| SC DHEC | S.C. Department of Health and Environmental Control |
| SDF | Saltstone Disposal Facility |
| SPF | Saltstone Production Facility |
| SRNL | Savannah River National Laboratory |
| SRS | Savannah River Site |
| SWPF | Salt Waste Processing Facility |
| TEDE | Total Effective Dose Equivalent |
| TSR | Technical Safety Requirements |
| USEPA | United States Environmental Protection Agency |
| USDOE | United States Department of Energy |
| USNRC | United States National Regulatory Commission |
| WSRC | Westinghouse Savannah River Company |

## 1.0 Executive Summary

The ability to safely process the salt component of the waste stored in the underground storage tanks at Savannah River Site (SRS) is a crucial prerequisite for completion of the overall SRS waste disposition plan. Failure to proceed expeditiously with salt processing will have many significant consequences on the Department of Energy's (DOE) efforts to accelerate reduction of the risks posed by the continued long-term storage of this waste within the aging infrastructure of the SRS Tank Farms. Removal and disposal of low-activity salt waste is critical in order to establish empty tank space for future tank waste processing operations including the Actinide Removal Process (ARP), Modular Caustic Side Solvent Extraction Unit (MCU) and the Salt Waste Processing Facility (SWPF) and to assure that vitrification of the high-activity fraction will be able to continue uninterrupted.

DOE contemplates removing fission products and actinides from the salt waste using a variety of technologies, combining the removed fission products and actinides with the metals being vitrified in the Defense Waste Processing Facility (DWPF), and solidifying the remaining low-activity salt stream into a grout matrix known as saltstone grout suitable for disposal in vaults at the Saltstone Disposal Facility (SDF) at SRS. The ability to dispose of the low-activity salt stream in the SDF requires determination of compliance to Section 3116 of the National Defense Authorization Act (NDAA) (*Ronald W. Reagan National Defense Authorization Act for FY 2005*, Section 3116, 2004.). One of the requirements of Section 3116 of the NDAA is to demonstrate compliance with the performance objectives set out in Subpart C of Part 61 of Title 10, Code of Federal Regulations (CFR) (USNRC, 2004).

This Performance Objective Demonstration Document (PODD) addresses the disposal of solidified low-activity salt waste streams into the SDF as saltstone grout and its compliance with performance objectives for near-surface disposal of radioactive waste. Specifically, this PODD demonstrates and documents that the solidified low-activity salt streams from the SRS salt processing activities meet the performance objectives set out in Subpart C of Part 61 of Title 10, Code of Federal Regulations.

The performance objectives from 10 CFR 61 and the results demonstrating compliance are as follows:

- 10 CFR 61.41 – The all-pathways dose-based performance objective is that no member of the public may receive an annual dose exceeding 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ. The evaluation of this performance objective is presented in Section 4. The calculated annual doses to a member of the public are 2.3 mrem to the whole body, 4.6 mrem to the thyroid and 5.3

mrem to any other organ. These doses are well within the 10 CFR 61.41 performance objective.

- 10 CFR 61.42 – The inadvertent intruder dose-based performance objective is to protect any member of the public from intruder scenarios. The 500 mrem to the whole body has been chosen from the basis for 10 CFR 61 waste classification discussed in Section 5.2 of the Final Environmental Impact Statement on 10 CFR 61 (USNRC, 1982). The evaluation of this performance objective is presented in Section 5. The calculated annual dose to the inadvertent intruder of 22 mrem to the whole body is well within the 10 CFR 61.42 performance objective.

- 10 CFR 61.43 – The operational radiation exposure performance objective is that operation of the SDF is conducted in compliance with 10 CFR 20 (USNRC, 2005) and 10CFR 61.41. (10 CFR 61.41 is addressed above) The evaluation of this performance objective is presented in Section 6. The occupational radiation protection program ensures compliance with the 10 CFR 61.43 performance objective.

- 10 CFR 61.44 – The disposal facility performance objective is that the design, operation and closure of the SDF will achieve long-term stability of the site to eliminate to the extent practicable the need for ongoing maintenance. The evaluation of this performance objective is presented in Section 7. The disposal facility design ensures compliance with the 10 CFR 61.44 performance objective.

Based upon the information presented in this PODD, the SRS low-activity salt stream to be disposed of in the SDF complies with all 10 CFR 61 Subpart C performance objectives for the disposal of SRS salt waste streams.

# 2.0 Introduction and Background

## 2.1 Salt Processing Strategy

Given that the SWPF cannot be constructed, permitted, and operated until approximately 2009, DOE developed a two-part interim processing approach that is tailored to the physical attributes of the different waste forms and the existing available technologies. This approach will allow ongoing risk reduction activities through sludge removal to continue at its current accelerated pace until SWPF is operational without resulting in back-end delays, thereby reducing risks to public health and safety, and the environment, as well as occupational risks. This approach also supports acceleration of SRS H-Canyon stabilization efforts and SWPF processing.

The Interim Salt Processing Strategy will be implemented so as to minimize the curies in the waste that will be disposed of at SDF following treatment using the interim technologies. This is accomplished by: 1) processing only that salt waste volume necessary to provide the minimum tank space needed to support efficient DWPF and SWPF operations and 2) carefully selecting the salt waste that will be processed from the tanks containing lower activity waste (with the majority of the total curies disposed of in SDF being associated with the shorter-lived Cs-137) (Mahoney and d'Entremont, 2004).

The Interim Salt Processing Strategy provides for treatment of these lower activity streams to remove the Cs-137 and other actinides with a lower contribution to total activity utilizing Deliquification, Dissolution, and Adjustment (DDA) and ARP/MCU technologies. (Appendix A provides a brief description of each of the interim processing technologies.) This tank selection process limits the total estimated radioactivity inventory permanently disposed of in SDF vaults to approximately 3 million curies (MCi) (Mahoney and Chew, 2004). Due to some uncertainty associated with the

characterization of the saltcake waste, the actual curie content of this volume of material may be as high as 5 MCi (Mahoney and Chew, 2004). Greater than 99% of the curies are associated with Cs-137, which has a half-life of approximately 30 years, and its equilibrium daughter product Ba-137m, which has a half-life of minutes.

After undergoing removal of radionuclides through DDA, ARP/MCU, and SWPF, as the case may be, the low-activity salt solution will be solidified into a grout matrix and disposed of in the SDF vaults at SRS. The low-activity salt solution will be mixed with dry chemicals (cement, slag, and flyash) to form a homogeneous grout mixture in the Saltstone Production Facility (SPF) and the slurry is transferred to SDF where it solidifies. SDF and SPF are collectively referred to as the Saltstone Facility.

## 2.2 Performance Objective Demonstration Document (PODD) Purpose

The ability to dispose of the low-activity salt stream in the SDF requires determination of compliance to Section 3116 of the NDAA. One of the requirements of Section 3116 of the NDAA is to demonstrate compliance with the performance objectives set out in Subpart C of Part 61 of Title 10, CFR. Section 3116 of the NDAA provides in relevant part:

[T]he term "high-level radioactive waste" does not include waste from reprocessing of spent nuclear fuel that the Secretary of Energy…in consultation with the Nuclear Regulatory Commission…determines –

***

(3)(A) [W]ill be disposed of –

(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations[.]

The purpose of this PODD is to demonstrate and document that the solidified low-activity salt streams from the SRS salt processing activities meet the performance objectives set out in Subpart C of Part 61 of Title 10, Code of Federal Regulations (USNRC, 2004). This PODD describes the process, analysis methods, input/assumptions, results and references necessary to demonstrate compliance with the performance objectives of 10 CFR 61.41, through 44 by inclusion of pertinent data, results, and linking to supporting documents, i.e. 2005 updated Special Analysis (Cook et al., 2005).

## 2.3 Subpart C of 10 CFR Part 61 Performance Objectives

Subpart C of 10 CFR Part 61 lists four performance objectives.

10 CFR 61.41 states:

> Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable.

The demonstration of 10 CFR 61.41 compliance is presented in Section 4.

10 CFR 61.42 states:

> Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed.

The demonstration of 10 CFR 61.42 compliance is presented in Section 5.

10 CFR 61.43 states:

> Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in Part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by Section 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.

The demonstration of 10 CFR 61.43 compliance is presented in Section 6. The concepts of As Low As Reasonably Achievable (ALARA) and reasonable assurance as applied to SDF are addressed in Sections 6.11 and 6.12, respectively.

10 CFR 61.44 states:

> The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.

The demonstration of 10 CFR 61.44 compliance is presented in Section 7.

### 2.4 Saltstone Disposal Facility Performance Evaluations

Over time, the Saltstone Facility has undergone revisions in the anticipated radiological inventory and the models used to evaluate compliance with the performance objectives. Thus, over the course of time, the performance objective compliance evaluations have been calculated in various documents to reflect new information and methodology. This section will describe the evolution of performance objective demonstrations and the process for the most recent demonstration of compliance with 10 CFR 61 Subpart C.

#### 2.4.1 1992 - Radiological Performance Assessment for the Z-Area Saltstone Disposal Facility

In 1992 during the design and construction of the Saltstone Facility, a Radiological Performance Assessment (PA) was produced (MMES, 1992) hereafter referred to as the 1992 PA. The 1992 PA was written against the requirements contained in DOE Order 5820.2A (USDOE, 1988a). The inventory assumption for this initial evaluation was the decontaminated salt solution stream from the In-Tank Precipitation Facility (ITP) and the concentrate stream from the Effluent Treatment Process (ETP). The 1992 PA evaluated waste material in SDF Vaults 1 and 4 and projected the results for future vaults. The groundwater evaluation point for this and future performance objective compliance reviews was a hypothetical well at a point 100 meters from the disposal vaults. The groundwater pathway objective assumed in the 1992 PA was a dose of 4 mrem from direct ingestion for all radionuclides combined. The groundwater pathway analysis was quantitative in nature while the air pathway was a qualitative evaluation which concluded that the air pathway was insignificant compared to the groundwater pathway. The intruder scenarios were evaluated in a qualitative fashion. The dose calculations in the PA were based on the anticipated inventory and determined to be within performance limits. An all-pathways evaluation was not completed.

#### 2.4.2 1998 - Addendum to the Radiological Performance Assessment for Z-Area

A 1998 Addendum was developed for the 1992 PA in order to address comments from the DOE Performance Assessment Peer Review Panel and DOE-HQ (WSRC, 1998). There was no change in the conclusions of the 1992 PA based upon the 1998 Addendum.

#### 2.4.3 2002 - Special Analysis: Reevaluation of the Inadvertent Intruder, Groundwater, Air, and Radon Analyses for the Saltstone Disposal Facility

In 2002 a Special Analysis (SA) was performed (Cook et al., 2002), hereafter referred to as the 2002 SA. The 2002 SA was performed in response to updated facility inventory information due to suspending ITP and replacing that waste stream with expected low curie salt solution feed. DOE Order 435.1 (USDOE, 1999) had also been promulgated

and the 2002 SA used this order for a compliance determination. Rather than using a specific radionuclide list for evaluation, as was used in the 1992 PA, a radionuclide screening evaluation was performed using an National Council on Radiation Protection and Measurements (NCRP) methodology (NCRP, 1996). The screening yielded one new radionuclide of interest, Np-237. Rather than calculating specific doses from a fixed radionuclide inventory, the 2002 SA calculated radionuclide inventory limits against specific objectives of 25 mrem/yr from all-pathways, 10 mrem/yr from the air pathway and the EPA Maximum Contaminant Levels for the groundwater pathway (USEPA, 2000). The 1992 PA doses for a specific radionuclide inventory were utilized to determine the individual radionuclide limits. For $\beta-\gamma$ (beta-gamma) radionuclides the limit is 4 mrem/yr from the direct ingestion groundwater pathway and 15 pCi/liter for the $\alpha$ (alpha) radionuclides. Since Np-237 was not included in the 1992 PA, the limit for this radionuclide was calculated in the 2002 SA. The air pathway is a quantitative analysis in the 2002 SA. The anticipated low curie salt solution feed inventory was then compared to the limits. The results of intruder analyses and groundwater and air pathways demonstrated compliance with DOE Order 435.1.

### 2.4.4 2005 - Special Analysis: Revision of Saltstone Vault 4 Disposal Limits

The latest information on the SDF feed solutions, updated modeling methods, updated closure cap design and evaluations are captured in a 2005 updated Special Analysis (Cook et al., 2005), hereafter referred to as the 2005 SA. The 2005 SA supplements the analyses in the 1992 PA and supersedes the analyses in the 2002 SA. The 2005 SA evaluates the SDF against the specific performance objectives of DOE 435.1 (USDOE, 1999) and 10 CFR 61.

The salt waste inventory was updated to reflect current estimates of the total inventory expected to be disposed of in Vault 4 (DDA and the anticipated decontaminated solution from the SWPF). The 2005 SA recalculates groundwater transport and air transport. An all-pathways analysis includes all residential and agricultural pathways. The radionuclide limits are calculated for specific exposure pathways and a limit determined for each pathway at the most restrictive exposure time period between 100 and 10,000 years after closure. The most restrictive limit for any pathway for each radionuclide is chosen as the limit and thus a sum-of-fractions for any waste inventory will be conservative. The 2005 SA includes many sensitivity evaluations but it only includes limits for the SDF Vault 4. The intruder scenarios have also been recalculated and demonstrate compliance with the 10 CFR 61.42 performance objective.

### 2.4.5 2005 – Future Vaults Evaluation

As indicated in Section 2.4.4, the 2005 SA only addresses the inventory anticipated for Vault 4. In order to demonstrate compliance for future SDF operations and inventory, the same methodology as the 2005 SA is utilized. The projected inventory for all SDF vaults has been determined from the sum of the present inventory (Crapse et al., 2004) and

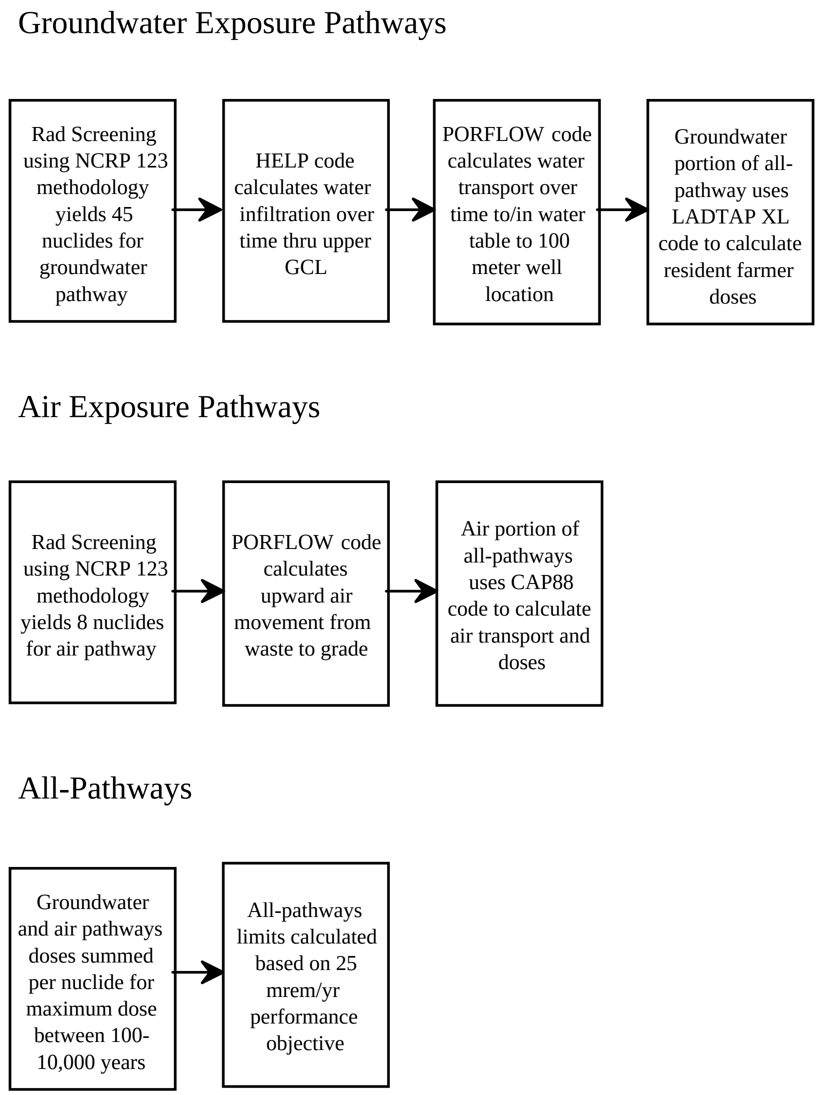
projected operations (d'Entremont and Drumm, 2005) and is presented in Section 3.2. For demonstration of compliance with 10 CFR 61.41 performance objectives, the projected inventory of the entire Saltstone facility has been conservatively assumed to all be contained in Vault 4. By putting the entire SDF projected inventory in Vault 4, versus spreading the inventory over many additional vaults, the plume diffusion and concentration dilution that would actually be present at a single 100 meter well location is not taken into consideration. For compliance of future vaults with the 10 CFR 61.42 intruder performance objectives, the future vault is bounded by the inventory in Vault 4. This is conservative because Cs-137 is the dominant radionuclide for intruder dose and the DDA stream, which has the highest Cs-137 concentration, will be located primarily in Vault 4. The compliance of future vaults with 10 CFR 61.43 and 61.44 performance objectives are not dependent on the specific vault inventory but are reliant on SRS polices and procedures governing worker dose controls and vault design.

## 2.5 Process Overview for 2005 SA Analyses

This section provides an overview of the analysis steps utilized as part of the 2005 SA process to demonstrate compliance with the 10 CFR 61 Subpart C performance objectives.

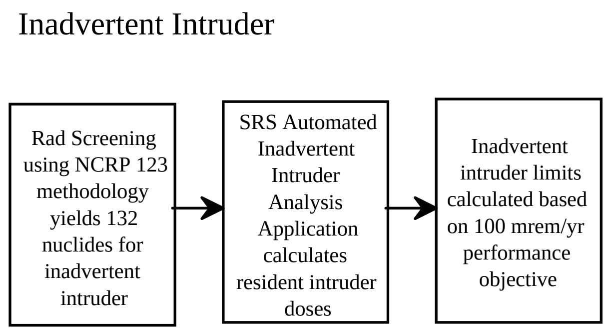
The evaluation to demonstrate compliance with the 10 CFR 61.41 all-pathways performance objective includes evaluations for groundwater exposure pathways and air exposure pathways. Figure 2-1 illustrates the analysis steps. The methods, input parameters and references, specific analysis codes and results for the 10 CFR 61.41 all-pathways evaluation are detailed in Section 4 of this PODD.

**Figure 2-1: Process Overview For All-Pathways Evaluation**




The evaluation to demonstrate compliance with the 10 CFR 61.42 inadvertent intruder performance objective includes a resident intruder. Figure 2-2 illustrates the analysis steps in simple terms. The methods, input parameters and references, specific analysis code and results for the 10 CFR 61.42 inadvertent intruder evaluation are detailed in Section 5 of this PODD.

**Figure 2-2: Process Overview For Inadvertent Intruder Evaluation**




The information necessary to demonstrate compliance with the occupational radiation exposure performance requirement of 10 CFR 61.43 is presented in Section 6 of this PODD detailing the SRS policies and procedures for ensuring worker radiological protection and controls.

The information necessary to demonstrate compliance with the SDF long-term stability performance requirement of 10 CFR 61.44 is presented in Section 7 of this PODD detailing the SDF design and closure concept to ensure long-term stability.

# 3.0 Saltstone Facility Description

The Saltstone Facility (located in the Z-Area of SRS) consists of two facility segments: the SPF, which receives and treats the salt solution to produce saltstone grout, and the Saltstone Disposal Facility (SDF), which consists of vaults used for the final disposal of the saltstone grout. Both the SPF and the SDF are located in Z-Area. The SPF is permitted as a wastewater treatment facility per the South Carolina Department of Health and Environmental Control (SCDHEC) Regulations R.61-67. Construction of SPF and the first two vaults of SDF were completed between February 1986 and July 1998. The Saltstone Facility started radioactive operations June 1990. Future vaults will be constructed on a "just-in-time" basis in coordination with salt processing production rates.

## 3.1 Saltstone Facility Physical Description

### 3.1.1 Physical Location Description

The Saltstone Facility is located approximately 6.2 miles from the nearest SRS site boundary. Z-Area, where the Saltstone Facility is located, lies on a local topographic high at approximately 290 feet above mean sea level (WSRC, 2004a). This site was selected for low-level waste disposal because of its location on a well-drained topographic high, as evidenced by the lack of marshes or other bodies of standing water. Z-Area is bounded by two streams: Upper Three Runs and McQueen Branch, a tributary to Upper Three Runs. McQueen Branch is located approximately 1.0 miles to the East and 0.75 miles to the Northeast of Z-Area, and Upper Three Runs is located approximately 1.0 miles to the Northwest (WSRC, 2004a). Under Z-Area, the minimum depth to the water table from the ground surface in any given year for the specific well locations studied is estimated to be 43 feet (i.e., at 247 feet above mean sea level) on the basis of water table fluctuations (Cook, 1983). Per South Carolina Department of Health and Environmental Control (SCDHEC) Regulation R.61-107.16, the bottom of all current and future disposal vaults must be located at least two feet above this historic high water table. This study has been recently extended to depict the probable maximum water table contours within Z-Area (Hiergesell 2005). The probable maximum contours will provide input to the design of future vaults to ensure that, even under extreme wet climatic conditions, the water table will never rise too close to the base of a vault.

There are no lakes or flow control structures on Upper Three Runs or its tributaries. The Probable Maximum Flood value for Upper Three Runs is 175 feet above mean sea level, which is substantially below the planned maximum depth of the SDF vaults (WSRC, 2004a). The 500-year and 100-year flood plains are located at 153.1 feet and 151.7 feet above mean sea level, respectively (Chen, 1999). An in-depth discussion of the hydrogeology of Z-Area is provided in Section 2.2 of the 1992 PA (MMES, 1992).

Figures 3-1 and 3-2 graphically represent the location of various streams and areas in relation to Z-Area.

**Figure 3-1. SRS Low Level Radioactive Waste Disposal Facilities**

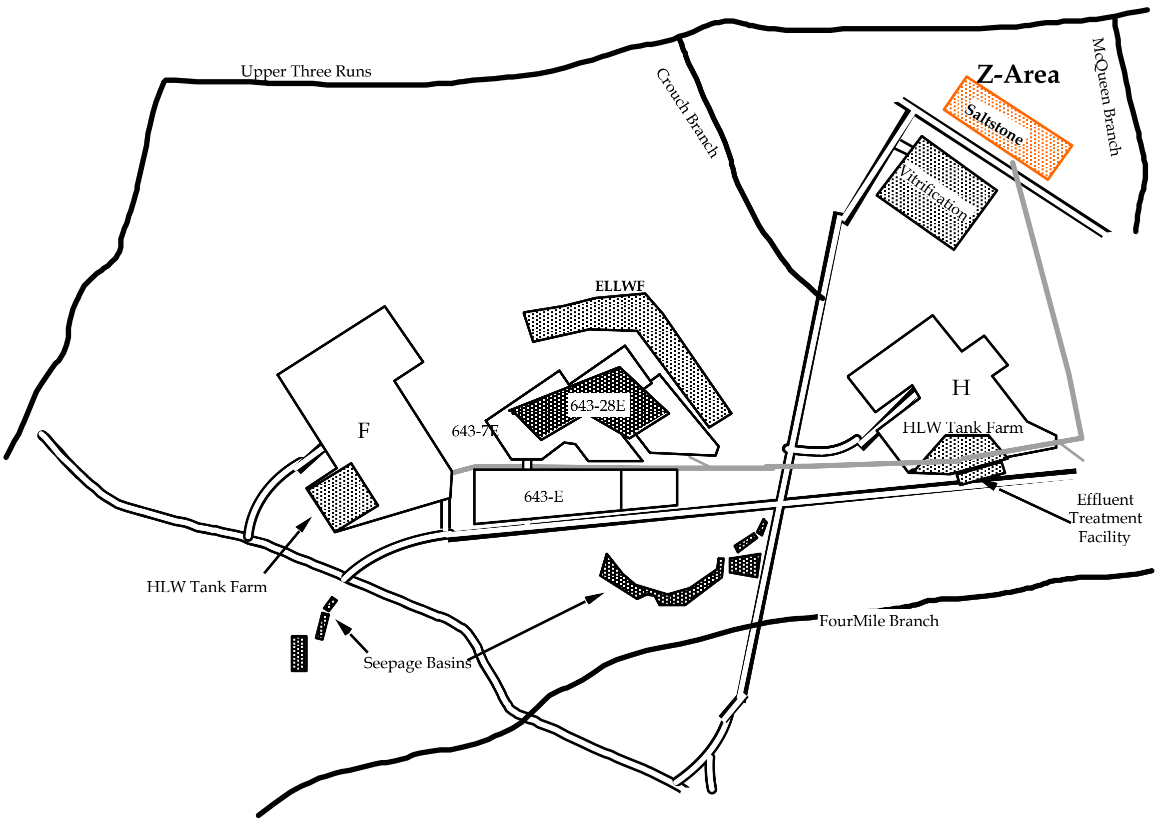
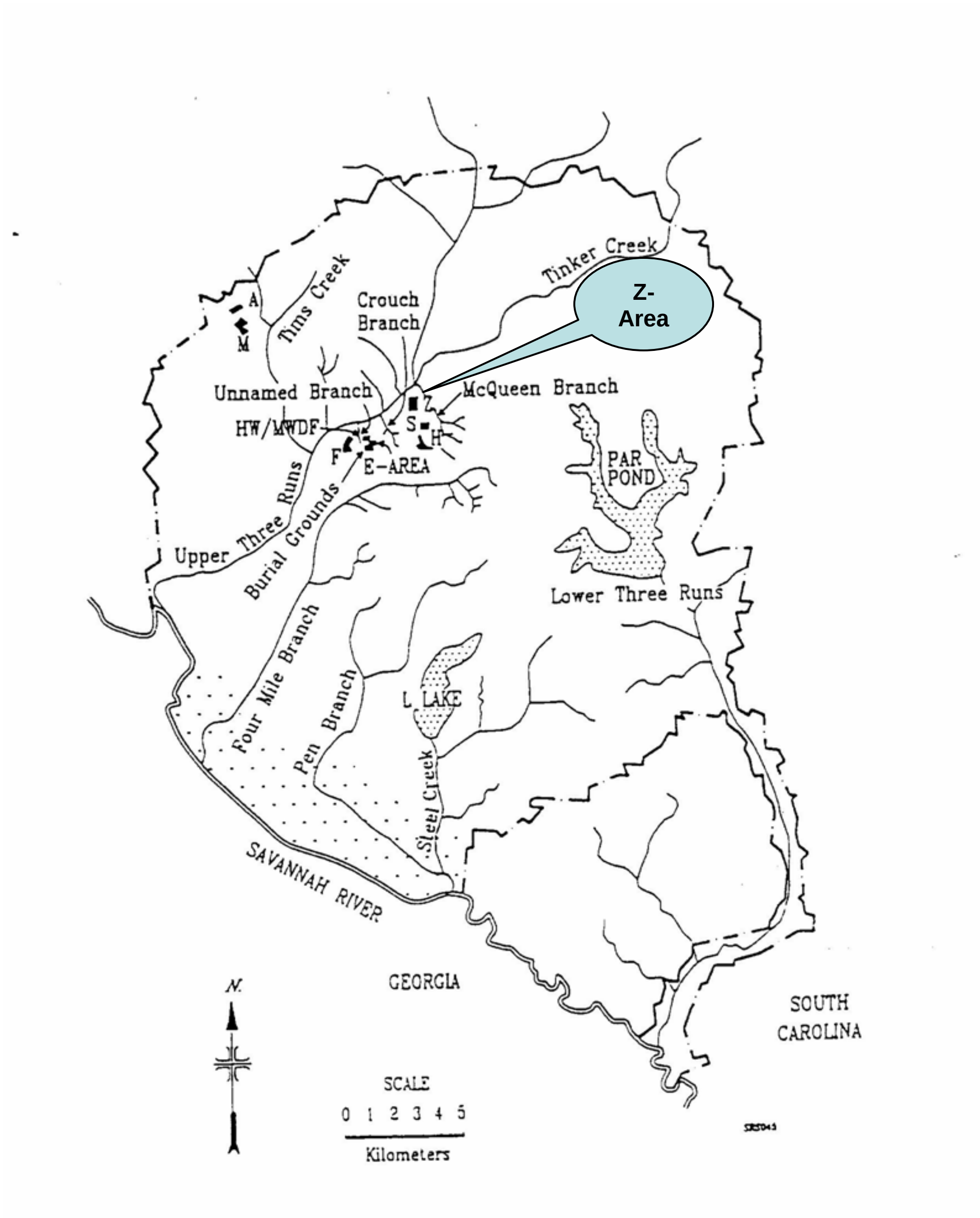


**Figure 3-2. Facility Location Map of SRS Showing Surface Drainage**



### 3.1.2 Saltstone Facility Vault Description

The two existing vaults (i.e., Vault 1 and Vault 4) are constructed of reinforced concrete containing slag (Langton, 1986) [Note that Vaults 2 and 3 have not yet been built]. Both Vaults 1 and 4 have been partially filled with previous saltstone grout pours. Vault 1 is not currently considered an active vault. The currently active vault (Vault 4) has the dimensions of approximately 200 feet wide, by 600 feet in length, by 26 feet in height. The vault is divided into 12 cells, with each cell measuring approximately 100 ft. x 100 ft. The vault is covered with a sloped, permanent roof that has a minimum thickness of 4 inches, and a minimum slope of 0.24 inches/foot. The vault clean concrete walls are approximately 1.5 feet thick and have a base mat with a thickness of 2 feet. Operationally, the cells of the vault will be filled to a height of approximately 25 feet with saltstone grout. Figure 3-3 is a view of SDF vaults. Table 3-1 provides the dimensions of SDF Vault 4.

**Figure 3-3. View of SDF Vault 4**

**Table 3-1. Dimensions of Saltstone Vault 4 (Cook et al., 2005)**

| Component | Dimensions of Vertical Distances | | |
|---|---|---|---|
| | From (ft) | To (ft) | Approximate Thickness (ft) |
| Native Soil | 0.00 | 40.00 | 40.00 |
| Bottom Concrete Slab | 40.00 | 42.00 | 2.00 |
| Saltstone | 42.00 | 66.75 | 24.75 |
| Concrete at Center[1] | 66.75 | 70.50 | 3.75 |
| Drainage Layer[2] | 70.50 | 72.50 | 2.00 |
| Drainage Layer at the Vault Base | 40.00 | 45.00 | 5.00 |
| Backfill above Drainage Layer[3] | 72.50 | 77.50 | 5.00 |
| | | | |
| | Dimensions of Horizontal Distances | | |
| Center Slab[4] | 0.00 | 0.75 | 0.75 |
| Saltstone | 0.75 | 99.25 | 98.50 |
| Side Slab | 99.25 | 100.75 | 1.50 |
| Drainage Layer | 100.75 | 103.75 | 3.00 |
| Drainage Layer at the Vault Base | 100.75 | 110.75 | 10.00 |

[1] Concrete includes tip of vault wall, concrete pour and concrete roof.
[2] Slope = 2.0%
[3] Slope = 3.0% at the upper boundary
[4] Actual center slab thickness = 1.50 ft.

### 3.1.3 SDF Closure Concept

The developed closure concept is illustrated in Figures 3-4 and 3-5. After an individual vault cell is filled with Saltstone, interim closure will be performed which consists of the placement of a clean grout layer between the saltstone grout and the overlying concrete roof. Final closure of the SDF will be accomplished by constructing a drainage system and revegetating the site. The drainage system will consist of a system of rip-rap lined ditches that intercept the gravel layer of the moisture barrier. These ditches will divert surface runoff and water intercepted by the moisture barrier away from the disposal site. The drainage ditches will be constructed between rows of vaults and around the perimeter of the SDF. The topsoil will be revegetated with bamboo.(Phifer and Nelson, 2003)

**Figure 3-4: Saltstone Vault Engineered Closure Cap**

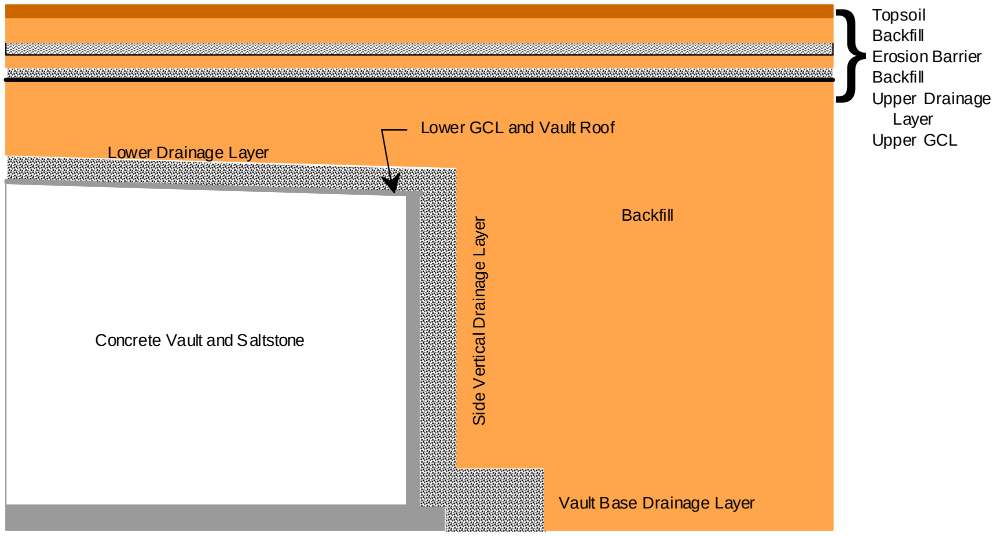
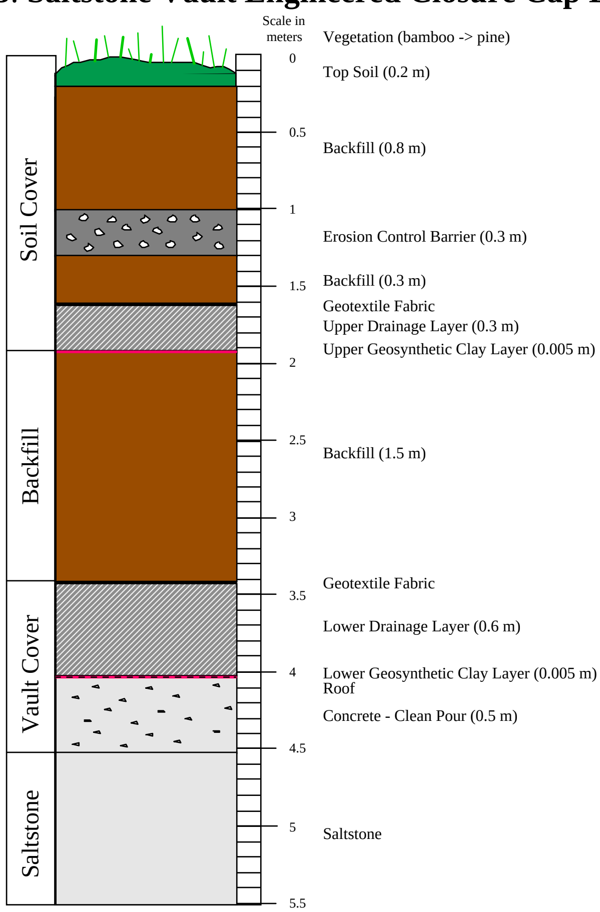



**Figure 3-5**: **Saltstone Vault Engineered Closure Cap Description**



## 3.2 Radiological Inventory

### 3.2.1 Projected Saltstone Inventory

The projected final inventory of the SDF combines the existing inventory of the partially filled vaults 1 and 4 (Crapse et al, 2004) and all projected future inventory additions, based on the present SRS salt processing strategy (d'Entremont and Drumm, 2005). The future inventory additions be added to existing Saltstone vaults as well as other vaults yet to be constructed Table 3-2 presents the projected total SDF inventory as calculated by these references.

**Table 3-2. Projected Total Saltstone Disposal Facility Inventory**

| Nuclide | Current Inventory Vault 1 (Ci) | Current Inventory Vault 4 (Ci) | Future Inventory Additions (Ci) | Total (Ci) |
|---|---|---|---|---|
| H-3 | 2.73E+01 | 2.94E+01 | 9.37E+03 | 9.43E+03 |
| C-14 | 1.28E+00 | 2.35E-01 | 5.18E+02 | 5.20E+02 |
| Na-22 | NV | NV | 5.05E+03 | 5.05E+03 |
| Al-26 | NV | NV | 2.35E+01 | 2.35E+01 |
| Ni-59 | 3.46E-02 | 9.09E-03 | 2.81E+00 | 2.85E+00 |
| Co-60 | 2.77E-03 | 6.83E-03 | 1.10E+02 | 1.10E+02 |
| Ni-63 | 9.38E-01 | 6.01E-02 | 2.50E+02 | 2.51E+02 |
| Se-79 | 3.02E-01 | 2.57E-02 | 8.91E+01 | 8.94E+01 |
| Sr-90 | 1.31E-02 | 3.17E-01 | 7.43E+03 | 7.43E+03 |
| Y-90 | NV | NV | 7.43E+03 | 7.43E+03 |
| Nb-94 | 2.51E-03 | 9.91E-04 | 7.23E-04 | 4.22E-03 |
| Tc-99 | 1.08E+02 | 2.35E+01 | 3.30E+04 | 3.31E+04 |
| Ru-106 | 1.14E-02 | 6.14E-03 | 2.28E+03 | 2.28E+03 |
| Rh-106 | NV | NV | 2.28E+03 | 2.28E+03 |
| Sb-125 | 1.29E+00 | 9.39E-01 | 9.24E+03 | 9.24E+03 |
| Te-125m | NV | NV | 2.26E+03 | 2.26E+03 |
| Sn-126 | 9.97E-01 | 5.66E-02 | 4.50E+02 | 4.51E+02 |
| Sb-126 | NV | NV | 6.30E+01 | 6.30E+01 |
| Sb-126m | NV | NV | 4.50E+02 | 4.50E+02 |
| I-129 | 1.12E-01 | 8.16E-02 | 1.78E+01 | 1.80E+01 |
| Cs-134 | NV | 1.32E-02 | 2.71E+03 | 2.71E+03 |
| Cs-135 | NV | NV | 4.67E+00 | 4.67E+00 |
| Cs-137 | 7.96E+00 | 1.68E+01 | 1.35E+06 | 1.35E+06 |
| Ba-137m | NV | NV | 1.28E+06 | 1.28E+06 |
| Ce-144 | NV | NV | 6.27E+00 | 6.27E+00 |
| Pr-144 | NV | NV | 6.27E+00 | 6.27E+00 |
| Pm-147 | NV | NV | 4.14E+03 | 4.14E+03 |
| Sm-151 | NV | 9.29E-04 | 4.55E+03 | 4.55E+03 |
| Eu-152 | 6.92E-03 | 5.14E-03 | 2.20E+01 | 2.20E+01 |

| Table 3-2. Projected Total Saltstone Disposal Facility Inventory | | | | |
|---|---|---|---|---|
| Nuclide | Current Inventory Vault 1 (Ci) | Current Inventory Vault 4 (Ci) | Future Inventory Additions (Ci) | Total (Ci) |
| Eu-154 | 2.01E-03 | 9.03E-03 | 9.74E+02 | 9.74E+02 |
| Eu-155 | NV | 1.58E-03 | 2.57E+02 | 2.57E+02 |
| Ra-226 | NV | NV | 1.30E+01 | 1.30E+01 |
| Ra-228 | NV | NV | 1.04E-01 | 1.04E-01 |
| Ac-227 | NV | NV | 1.91E-05 | 1.91E-05 |
| Th-229 | NV | NV | 7.53E-03 | 7.53E-03 |
| Th-230 | NV | NV | 3.53E-02 | 3.53E-02 |
| Pa-231 | NV | NV | 5.32E-05 | 5.32E-05 |
| Th-232 | NV | NV | 1.04E-01 | 1.04E-01 |
| U-232 | NV | 9.46E-03 | 2.14E-02 | 3.09E-02 |
| U-233 | NV | NV | 2.22E+00 | 2.22E+00 |
| U-234 | 2.85E-01 | 3.52E+00 | 3.91E+00 | 7.72E+00 |
| U-235 | 3.17E-03 | 6.81E-02 | 6.38E-02 | 1.35E-01 |
| U-236 | NV | NV | 3.03E-01 | 3.03E-01 |
| U-238 | 7.36E-03 | 1.10E-01 | 5.07E+00 | 5.19E+00 |
| Np-237 | 4.49E-03 | 4.87E-03 | 2.11E+00 | 2.12E+00 |
| Pu-238 | 9.63E-03 | 6.78E-01 | 1.36E+04 | 1.36E+04 |
| Pu-239 | 1.23E-02 | 1.33E-01 | 6.55E+02 | 6.55E+02 |
| Pu-240 | NV | NV | 1.75E+02 | 1.75E+02 |
| Pu-241 | 3.59E-02 | 1.63E-01 | 7.03E+03 | 7.03E+03 |
| Pu-242 | 9.03E-04 | 8.03E-03 | 1.72E-01 | 1.81E-01 |
| Am-241 | 4.92E-04 | 6.67E-02 | 9.49E+01 | 9.50E+01 |
| Am-242m | NV | NV | 5.27E-02 | 5.27E-02 |
| Pu-244 | NV | NV | 7.96E-04 | 7.96E-04 |
| Am-243 | NV | 1.30E-03 | 2.05E-02 | 2.18E-02 |
| Cm-242 | NV | NV | 1.05E-01 | 1.05E-01 |
| Cm-243 | NV | NV | 2.67E-02 | 2.67E-02 |
| Cm-244 | NV | 8.06E-02 | 8.71E+01 | 8.72E+01 |
| Cm-245 | NV | NV | 8.58E-03 | 8.58E-03 |
| Cm-247 | NV | NV | 5.15E-12 | 5.15E-12 |
| Cm-248 | NV | NV | 5.36E-12 | 5.36E-12 |
| Bk-249 | NV | NV | 6.31E-19 | 6.31E-19 |
| Cf-249 | NV | NV | 4.79E-11 | 4.79E-11 |
| Cf-251 | NV | 2.47E-01 | 1.64E-12 | 2.47E-01 |
| Cf-252 | NV | NV | 5.32E-14 | 5.32E-14 |
| | | | Total | 2.74E+06 |

Notes: NV indicates no value reported in reference

# 4.0 10 CFR 61.41 Compliance (All-Pathways Analysis)

10 CFR 61.41 states:

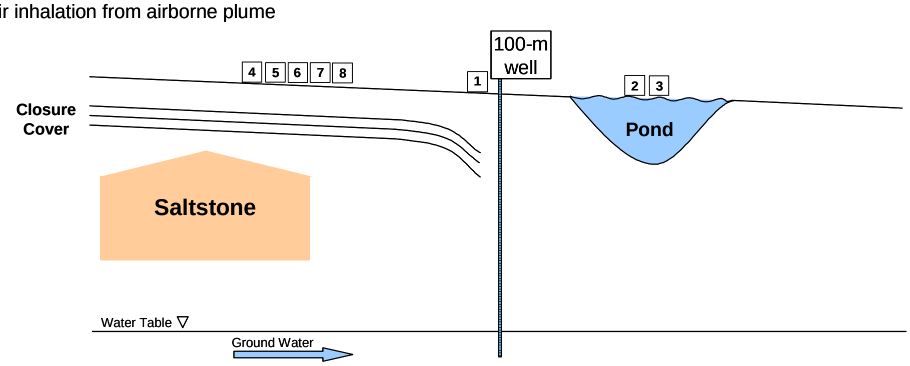
> Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable.

The all-pathways analysis is a combination of the doses derived from the maximum exposure via groundwater pathways and the doses from the maximum exposure via air pathways. Section 4.1 addresses the groundwater pathways analysis, Section 4.2 the air pathways analysis and Section 4.3 combines the two for the all-pathways analysis. See Figure 4-1 for the modeled pathways.

**Figure 4-1. Modeled 10 CFR 61.41 Pathways**

MEMBER OF PUBLIC

1. Drinking water from hypothetical well 100-meters from vault
2. Ingestion of fish from pond filled with water from 100-meter well
3. Direct irradiation resulting from recreational activities (swimming, boating, and shoreline use) in pond filled with water from 100-meter well
4. Ingestion of produce irrigated with water from 100-meter well and contaminated from the airborne plume
5. Ingestion of meat from cows fed vegetation irrigated with water from 100-meter well and contaminated from the airborne plume
6. Ingestion of milk from cows fed vegetation irrigated with water from 100-meter well and contaminated from the airborne plume
7. Direct irradiation from airborne plume
8. Air inhalation from airborne plume




[Not to Scale]

## 4.1 Groundwater Pathways Analysis

### 4.1.1 Groundwater Pathways Methodology

The groundwater pathway analysis for each radionuclide involves two steps. First a vadose zone flow and transport simulation is done to estimate flux to the water table for a disposed radionuclide parent and any subsequent progeny. Then saturated zone flow and transport modeling is used to estimate the groundwater concentration(s) at a hypothetical well placed 100 meters down-gradient from the disposal unit (USDOE, 2001b). Based on the facility description in section 3.1.1, Z-Area is not effected by groundwater contaminants from other facilities.

For radionuclides transported by the groundwater, the maximum groundwater concentration of each radionuclide within the time frame of interest (i.e., 10,000 years) (USNRC, 2000) is calculated and is input to the LADTAP XL© program (Jannik, 2005), which is a model used at SRS for demonstrating groundwater pathway dose compliance (Simpkins, 2004b). The maximum groundwater concentrations are calculated for a unit curie inventory of each radionuclide.

It is conservatively assumed that a future resident farmer uses the contaminated groundwater from a well 100 meters from the edge of Vault 4 (hereafter referred to as the 100 meter well) as a source of 1) drinking water, 2) pond water (in which fish are raised and recreational activities occur), and 3) irrigation water used for raising vegetables, meat, and milk. See Figure 4-1 for a pictorial of the modeled groundwater pathways.

### 4.1.2 Groundwater Pathway Radionuclide Screening

A screening analysis was conducted to produce a list of radionuclides requiring a detailed analysis for the groundwater pathway (Cook and Wilhite, 2004). The screening analysis used a methodology developed by the NCRP (NCRP, 1996) to determine the list of radionuclides of interest for the groundwater pathways. The 826 radionuclides originally considered potentially significant were reduced to 45 to be evaluated in detail. The 45 radionuclides and their daughter products are presented in Table 4-1 (Cook et al., 2005).

**Table 4-1. List of Modeled Radionuclides and Decay Daughters**

| | | | | | |
|---|---|---|---|---|---|
| Al-26 | Am-241 | Nb-94 | Np-237 | Ra-226 | U-235 |
| Am-243 | Np-237 | Nb-95m | *Pu-242 | Pb-210 | Pa-231 |
| Np-239 | Cm-246 | Nb-95 | *Pu5-242 | Po-210 | Ac-227 |
| Pu-239 | Cm-247 | Ni-59 | U-238 | Th-232 | Th-227 |
| Pu5-239 | Am-243 | Np-237 | *Pu-244 | Ra-228 | Ra-223 |
| Bi-210 | Np-239 | Pd-107 | *Pu5-244 | Th-228 | U-236 |
| Po-210 | Pu-239 | *Pu-238 | Ra-226 | Ra-224 | U-238 |
| C-14 | Pu5-239 | *Pu5-238 | Rb-87 | U-232 | Th-234 |
| Cf-249 | Cm-248 | U-234 | Se-79 | Th-228 | U-234 |
| Cm-245 | Pu-244 | *Pu-239 | Sn-126 | Ra-224 | Zr-93 |
| Pu-241 | Pu5-244 | *Pu5-239 | Sr-90 | U-233 | Nb-93m |
| Pu5-241 | Cs-135 | U-235 | Tc-99 | Th-229 | Zr-95 |
| Am-241 | Cs-137 | *Pu-240 | Th-228 | Ra-225 | Nb-95 |
| Np-237 | H-3 | *Pu5-240 | Ra-224 | U-234 | |
| Cl-36 | I-129 | U-236 | Th-229 | Th-230 | |
| Cm-245 | K-40 | *Pu-241 | Ra-225 | Ra-226 | |
| Pu-241 | Mo-93 | *Pu5-241 | Ac-225 | Pb-210 | |
| Pu5-241 | Nb-93m | Am-241 | Th-230 | Po-210 | |

Note: Left justified radionuclides are parent isotopes and indented radionuclides are their decay daughters.

* To indicate the plutonium oxidation states that were considered, Pu- represents the III,IV oxidation states and Pu5- represents the V,VI oxidation states

### 4.1.3 Groundwater Pathway Analysis Computation Code

This section describes the methods and mathematical codes used to evaluate the groundwater exposure pathways. This evaluation will demonstrate that the modeling used is adequate to ensure compliance with the requirements of 10 CFR 61.41 for the groundwater pathway. A three tiered approach was used to ensure predicted results are as accurate as possible. The approach used is:

- The Hydrologic Evaluation of Landfill Performance (HELP) code (USEPA, 1994a; USEPA, 1994b) is used as an analysis tool for water balance calculations through the upper Geosynthetic Clay Layer (GCL) (see Figure 3-4). The HELP model was developed by the U.S Army Corp of Engineers for the U. S. Environmental Protection Agency (EPA) to conduct landfill water balance analysis. It provides estimates of runoff, evapotranspiration, lateral drainage, vertical percolation, hydraulic head and water storage. Version 3.07 is the latest version of the model available from the EPA. In general, the HELP model

requires the following data, some of which may be selected from the default values.

1. Units
2. Location
3. Weather data file names
4. Evapotranspiration information
5. Precipitation data
6. Temperature data
7. Solar radiation data
8. Soil and design data file name
9. General landfill and site information
10. Landfill profile and soil/waste/geomembrane data
11. Soil Conservation Service runoff curve number information

- PORFLOW is a widely used commercially available flow and transport model that is used to calculate the fluxes of contaminants to the saturated zone and the concentration of contaminants in the saturated zone. It is a comprehensive software package that can analyze a wide range of environmental applications in flow and mass transport in geologic media. It provides for coupled transport of flow and multiple chemical species in complex three-dimensional geometry, transient or steady-state flow, confined or unconfined aquifers, fully or partially saturated media, single or multiple phase systems, and phase change between liquid and solid and liquid and gaseous phases. PORFLOW has been used extensively by the DOE, U. S. Geological Survey, U. S. Nuclear Regulatory Commission (NRC), U.S. Army and many commercial organizations (ACRI, 2002).

- LADTAP XL© (Simpkins, 2004a) is a spreadsheet developed by the Savannah River National Laboratory (SRNL) for use in estimating dose to individuals and populations resulting from the releases of radioactive materials via the groundwater pathway implementing the NRC LADTAP code methodology (USNRC, 1977b). The XL version is comprised of the two worksheets LADTAP and IRRIDOSE. LADTAP estimates the dose for environmental pathways including external exposures resulting from recreational activities, ingestion of water, fish and invertebrates. IRRIDOSE estimates the dose to the population from crop irrigation.

#### 4.1.3.1 Groundwater Pathway Quality Assurance for Modeling Codes

Appendix D describes the Quality Assurance (QA) Program implemented by WSRC. The QA program was conducted by modelers throughout the process utilizing established procedures for both engineering and research projects (SRS Procedure E7

Conduct of Engineering Manual, Procedure 2.60, Technical Reviews and Procedure 2.31 Engineering Calculations).

Software QA is conducted in accordance with the requirements of the WSRC 1Q Manual through the development and execution of Software Quality Assurance Plans. This procedure fulfills the requirements of DOE Order O414.1A, *Quality Assurance*, 10 Code of Federal Regulations (CFR) 830 Subpart A, “Quality Assurance Requirements”

- HELP has undergone substantial verification through the EPA. (See Appendix D) Enhancements to the model have been made to improve predictions and consequently the current version of HELP has been accepted by EPA and the regulated community as an appropriate tool for estimating water balance at landfills.

- The PORFLOW Software Quality Assurance Plan (Collard, 2002) describes the controls for the software and presents results of software grading and testing and acceptance results. Additionally the software vendor states that over 100 publications and project reports on the benchmarking, verification and application of the model are available. (See Appendix D for further description)

- LADTAP verification has been conducted to ensure results are consistent with expected results. Description of the verification process and results are contained in the User’s Manual. (Simpkins, 2004a)

#### 4.1.4 Groundwater Pathway Simulation Input Parameters and Assumptions

The dose estimates for the SDF were generated from the modeling efforts contained in the 2005 SA (Cook et al., 2005). The 2005 SA documents improvements in modeling methods, closure cap design and updated saltstone inventory due to facility changes. The 2005 SA recalculates both groundwater and air transport.

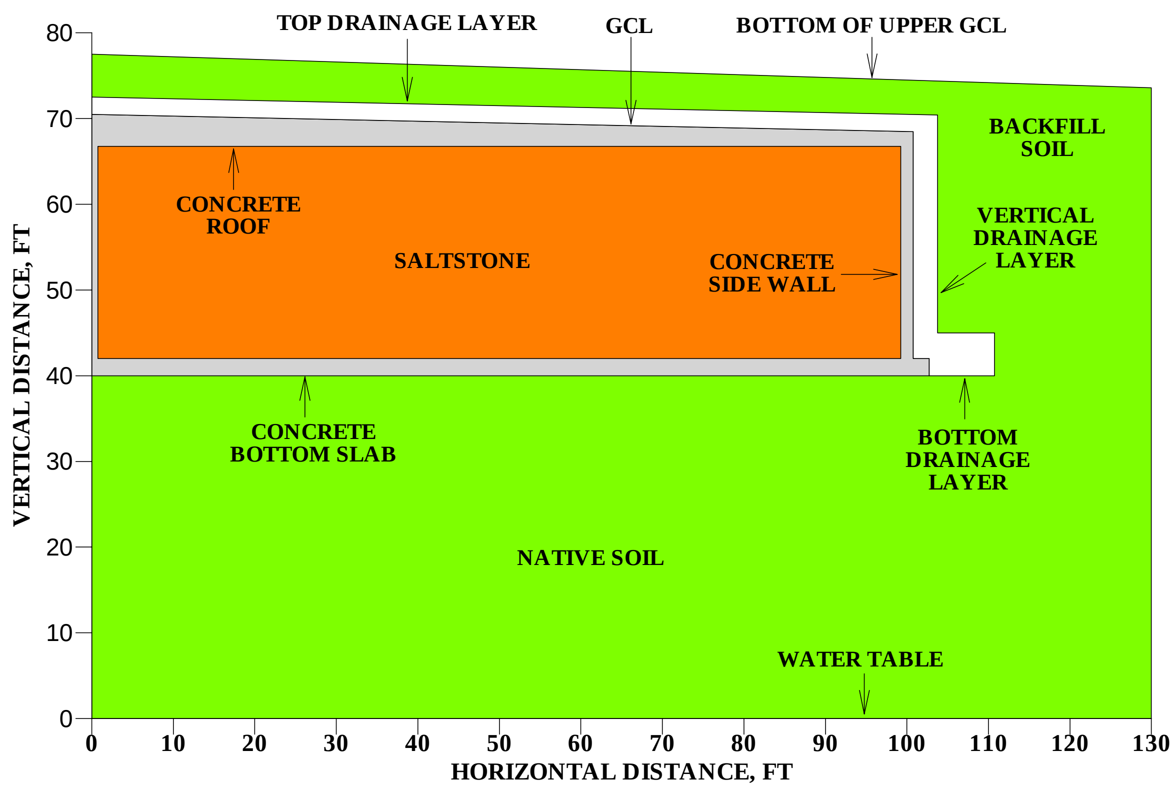
This section provides the inputs to the three simulation models used to generate the dose calculations and exposure pathways from contaminants contained in the groundwater.

The 2005 SA describes the evaluation models in the following manner. The dimensions of the vault and lower portion of the closure are summarized in Table 3-1. The “concrete” zone above the saltstone grout pour level (at 66.75 ft) includes the top portion of the center and exterior walls and the concrete roof. The drainage layer is a gravel/sand mixture. It is used to reduce water perching above the vault. Test modeling results indicate that perching water can increase water flow rate through the vault, which results in a higher contaminant leaching rate. The drainage layer is divided into three sections: top, vertical and bottom. The initial hydraulic conductivities in these sections are the same. However, these conductivities degrade at different rates (Phifer, 2004) as will be

described later. Because the backfill is largely soil excavated during vault construction, it is assumed that the backfill soil has the same properties as the native soil. There is a GCL above the vault roof. Since the hydraulic conductivity of the saltstone grout and the vault is less than or equal to the conductivity of the GCL ($10^{-9}$ cm/sec), the lower GCL is ignored in the simulation.

The HELP model was used to simulate water movement in the unsaturated zone. The modeling domain is defined as the bottom of the upper GCL to the top of the saturated zone. The PORFLOW simulates transport from the bottom of the upper GCL to the water table and to the compliance point. Figure 4-2 shows the domain of the PORFLOW simulation.

**Figure 4-2. Conceptual Model for Saltstone Vault 4 (Cook et al., 2005)**




It should be noted that only one half of the vault in the short dimension was modeled to take advantage of symmetry in the long dimension.

The hydraulic conductivities used in the simulation for the engineered porous media (saltstone grout, concrete, and gravel drain layers) were measured by Core Lab (Yu et al., 1993). These measured values are used for the first 100 years. The hydraulic conductivities are assumed to increase for saltstone grout and concrete in the ensuing

years. Table 4-2 shows the time periods used in the model as well as the assumptions in changing conditions of the cap that defines the time period. For instance, institutional controls for the first 100 years are assumed to maintain the cap intact. Phifer (2003) discusses in Chapters 3 and 4 the degradation over time. The mechanisms assumed to impact the hydraulic properties of the closure cap are pine forest succession, erosion and colloidal clay migration. These degradation mechanisms are the subject of a 2003 evaluation (Phifer, 2003). The changes in hydraulic properties as the closure cap degrades are presented in Table 4-3.

**Table 4-2. Material Property Results for HELP Modeling (Phifer, 2003)**

| Year | Vegetation | Topsoil Layer Thickness (inches) | Erosion Barrier Saturated Hydraulic Conductivity (cm/s) | Middle Backfill Layer Saturated Hydraulic Conductivity (cm/s) |
|---|---|---|---|---|
| 0 | Bamboo | 6 | 3.97E-04 | 1.00E-04 |
| 100 | Bamboo | 5.980 | 3.97E-04 | 1.20E-04 |
| 300 | Pine Forest | 5.940 | 3.98E-04 | 1.60E-04 |
| 550 | Pine Forest | 5.890 | 3.99E-04 | 2.30E-04 |
| 1,000 | Pine Forest | 5.800 | 4.01E-04 | 4.60E-04 |
| 1,800 | Pine Forest | 5.640 | 4.06E-04 | 1.60E-03 |
| 3,400 | Pine Forest | 5.320 | 4.15E-04 | 3.20E-03 |
| 5,600 | Pine Forest | 4.880 | 4.27E-04 | 3.20E-03 |
| 10,000 | Pine Forest | 4.0 | 4.51E-04 | 3.20E-03 |
| Year | Upper Drainage Layer Saturated Hydraulic Conductivity (cm/s) | One Square Centimeter Holes in Upper GCL[1] (#/acre) | Lower Drainage Layer Saturated Hydraulic Conductivity (cm/s) | |
| 0 | 1.00E-01 | 0 | 1.00E-01 | |
| 100 | 8.60E-02 | 0 | 1.00E-01 | |
| 300 | 6.30E-02 | 7,432 | 9.98E-02 | |
| 550 | 4.30E-02 | 26,013 | 9.89E-02 | |
| 1,000 | 2.10E-02 | 59,458 | 9.61E-02 | |
| 1,800 | 6.30E-03 | 118,916 | 8.96E-02 | |
| 3,400 | 3.20E-03 | 237,832 | 7.56E-02 | |
| 5,600 | 3.20E-03 | 401,341 | 5.62E-02 | |
| 10,000 | 3.20E-03 | 728,360 | 1.74E-02 | |

[1] Number of HELP model installation defects

**Table 4-3. Saturated Hydraulic Conductivities (cm/sec) (Cook et al., 2005)**

| Time(years) | 0 to 100 | 100 to 300 | 300 to 550 | 550 to 1,000 | 1000 to 1,800 | 1,800 to 3,400 | 3,400 to 5,600 | 5,600 to 10,000 |
|---|---|---|---|---|---|---|---|---|
| | | | | Horizontal conductivity: | | | | |
| Nati/Back | 1.00E-04 | 1.00E-04 | 1.00E-04 | 1.00E-04 | 1.00E-04 | 1.00E-04 | 1.00E-04 | 1.00E-04 |
| Drain Bot | 1.00E-01 | 9.99E-02 | 9.97E-02 | 9.90E-02 | 9.71E-02 | 9.30E-02 | 8.63E-02 | 7.46E-02 |
| Drain Ver | 1.00E-01 | 1.00E-01 | 1.00E-01 | 1.00E-01 | 1.00E-01 | 1.00E-01 | 1.00E-01 | 1.00E-01 |
| Drain Top | 1.00E-01 | 9.99E-02 | 9.93E-02 | 9.75E-02 | 9.28E-02 | 8.25E-02 | 6.58E-02 | 3.66E-02 |
| Concrete | 1.00E-12 | 5.20E-12 | 1.29E-11 | 3.16E-11 | 7.64E-11 | 1.98E-10 | 4.19E-10 | 1.00E-09 |
| Saltstone | 1.00E-11 | 3.00E-11 | 5.50E-11 | 1.00E-10 | 1.80E-10 | 3.40E-10 | 5.60E-10 | 1.00E-09 |
| | | | | Vertical conductivity: | | | | |
| Drain Bot | 9.52E-02 | 6.45E-02 | 2.70E-02 | 8.94E-03 | 3.34E-03 | 1.41E-03 | 7.25E-04 | 3.93E-04 |
| Drain Top | 8.89E-02 | 4.21E-02 | 1.29E-02 | 3.78E-03 | 1.36E-03 | 5.69E-04 | 2.91E-04 | 1.57E-04 |

The GCL has a saturated hydraulic conductivity of 5.0E-09 cm/sec as quoted by the manufacturer (GSE, 2002).

As time progresses fines intrude into the gravel layer and will begin to plug the drainage layer from the bottom. The cumulative amount of pluggage (Phifer, 2004) is estimated in Table 4-4. Until complete pluggage occurs (post 10,000 years) there will be no ponding and subsequent hydraulic pressure on the waste form. Therefore cracks in the waste form can be ignored due to adequate suction head in the waste form as indicated in Appendix A.4 of Cook et al., (2005).

**Table 4-4. Plugged-Zone Thickness as a Function of Time (Phifer, 2004;Freeze and Cherry, 1979)**

| Time (years) | Plugged-Zone Thickness (feet) |
|---|---|
| 0 | 0 |
| 100 | 0.0005 |
| 300 | 0.005 |
| 550 | 0.022 |
| 1,000 | 0.08 |
| 1,800 | 0.21 |
| 3,400 | 0.49 |
| 5,600 | 0.88 |
| 10,000 | 1.66 |

The initial hydraulic conductivity is $10^{-1}$cm/sec for the gravel zone.

The lower GCL is expected to degrade over time impacting the infiltration rate through the layer.  Table 4-5 (Phifer, 2004) presents the rates used in the simulation.

**Table 4-5. Infiltration Rates Used as Upper Boundary Conditions (Phifer, 2004)**

| Time Interval | Infiltration Rate (in/yr) |
|---|---|
| 0 to 100 | 0.39 |
| 100 to 300 | 1.73 |
| 300 to 550 | 5.48 |
| 550 to 1,000 | 9.97 |
| 1,000 to 1,800 | 12.90 |
| 1,800 to 3,400 | 13.90 |
| 3,400 to 5,600 | 14.06 |
| 5,600 to 10,000 | 14.09 |

The molecular diffusion coefficients chosen for use in the PORFLOW model are within the range reported for ionic solutes in porous media.  The values are near the end of the lower range for concrete and saltstone grout.  Table 4-6 presents the coefficients used in the modeling.

**Table 4-6. Molecular Diffusion Coefficients (Domenico and Schwartz, 1990)**

| Porous Media | $cm^2$/sec | $cm^2$/year |
|---|---|---|
| Native/Backfill Soil | 5.E-05 | 1.58E+02 |
| Drainage Layer | 5.E-05 | 1.58E+02 |
| Saltstone | 5.E-09 | 1.58E-01 |
| Concrete | 1.E-08 | 3.15E-01 |

Climate changes and cycles were evaluated. Data necessary for inclusion in the HELP model was synthesized using data from SRS Weather Stations. The data used in the model is presented in Table 4-7 and 4-8.

**Table 4-7. Average Monthly SRS Temperature and Precipitation Data (Phifer and Nelson, 2003)**

| Month | Average Temperature ($^oF$) | Average Precipitation (inches) |
|---|---|---|
| January | 46.3 | 4.38 |
| February | 50.0 | 3.95 |
| March | 57.2 | 4.68 |
| April | 64.3 | 2.91 |
| May | 72.1 | 3.56 |
| June | 78.4 | 4.99 |
| July | 81.6 | 5.43 |
| August | 80.3 | 5.41 |
| September | 75.2 | 3.93 |
| October | 65.1 | 3.12 |
| November | 56.7 | 2.96 |
| December | 48.8 | 3.45 |

**Table 4-8. Synthetic Daily Temperature and Precipitation Statistics over 100 Years (Phifer and Nelson, 2003)**

| | Average | Median | Standard Deviation | Minimum | High |
|---|---|---|---|---|---|
| Daily Temperature ($^oF$) | 64.73 | 66.50 | 14.24 | 19.40 | 92.70 |
| Yearly Temperature ($^oF$) | 64.73 | 64.69 | 0.83 | 62.40 | 66.89 |
| Daily Precipitation (inches) | 0.13 | 0.00 | 0.37 | 0.00 | 6.87 |
| Yearly Precipitation (inches) | 48.96 | 48.83 | 7.74 | 29.28 | 68.99 |

The HELP model employs many input parameters. Tables 4-9 and 4-10 present several of the site specific input parameters necessary.

**Table 4-9. Site Specific Input Parameters-Area and Initial Moisture for the HELP Model (Phifer, 2003)**

| Input Parameter (HELP Model Query) | Generic Input Parameter Value |
|---|---|
| Landfill area | 19.63 acres |
| Percent of area where runoff is possible | 100% |
| Do you want to specify initial moisture storage? (Y?N) | Y |
| Amount of water or snow on surface | 0 in. |

**Table 4-10. Site Specific Input Parameter Values-Curve Numbers (Phifer, 2003)**

| CN Input Parameter (HELP Model Query) | CN Input Parameter Value |
|---|---|
| Slope | 3% |
| Slope length | 450 ft. |
| Soil Texture | 5 (HELP model default soil texture) |
| Vegetation | 4 (i.e., a good stand of grass) |
| HELP Model Computed Curve Number | 54.4 |

Table 4-11 provides the actual HELP modeling values used for the initial time of 0 years. HELP model input parameters for subsequent years are found in Phifer (2003) through 10,000 years. Each of these cases accounts for an increasing degradation of the cap.

**Table 4-11. HELP Model Input Data (0 Years) (Phifer, 2003)**

**Input Date:**

| Input Parameter (HELP Model Query) | Generic Input Parameter Value |
|---|---|
| Landfill area = | 19.63 acres |
| Percent of area where runoff is possible = | 100% |
| Do you want to specify initial moisture storage? (Y/N) | Y |
| Amount of water or snow on surface = | 0 inches |
| CN Input Parameter (HELP Model Query) | CN Input Parameter Value |
| Slope = | 3% |
| Slope length = | 450 ft. |
| Soil Texture = | 5 (HELP model default soil texture) |
| Vegetation = | 4 (i.e., a good stand of grass) |
| HELP Model Computed Curve Number = 54.4 | |

| Layer | Layer Number | Layer Type |
|---|---|---|
| Topsoil | 1 | 1 (vertical percolation layer) |
| Upper Backfill | 2 | 1 (vertical percolation layer) |
| Erosion Barrier | 3 | 1 (vertical percolation layer) |
| Middle Backfill | 4 | 1 (vertical percolation layer) |
| Upper Drainage Layer | 5 | 2 (lateral drainage layer) |
| Upper GCL | 6 | 3 (barrier soil liner) |
| Lower Backfill | 7 | 1 (vertical percolation layer) |
| Lower Drainage Layer | 8 | 2 (lateral drainage layer) |
| Lower GCL | 9 | 3 (barrier soil liner) |

| | Layer Type | Layer Thickness (in) | Soil Texture No. | Total Porosity (Vol/Vol) | Field Capacity (Vol/Vol) | Wilting Point (Vol/Vol) | Initial Moisture (Vol/Vol) |
|---|---|---|---|---|---|---|---|
| 1 | 1 | 6 | - | 0.4 | 0.11 | 0.058 | 0.11 |
| 2 | 1 | 30 | - | 0.37 | 0.24 | 0.136 | 0.24 |
| 3 | 1 | 12 | - | 0.06 | 0.056 | 0.052 | 0.056 |
| 4 | 1 | 12 | - | 0.37 | 0.24 | 0.136 | 0.24 |
| 5 | 2 | 12 | - | 0.38 | 0.08 | 0.013 | 0.08 |
| 6 | 3 | 0.2 | - | 0.75 | 0.747 | 0.40 | 0.75 |
| 7 | 1 | 58.57 | - | 0.37 | 0.24 | 0.136 | 0.24 |
| 8 | 2 | 24 | - | 0.38 | 0.08 | 0.013 | 0.08 |
| 9 | 3 | 0.2 | - | 0.75 | 0.747 | 0.40 | 0.75 |

The lack of values in the table for particular parameters in particular layers denotes that no HELP model input was required for that parameter in that layer. No data are missing from the table

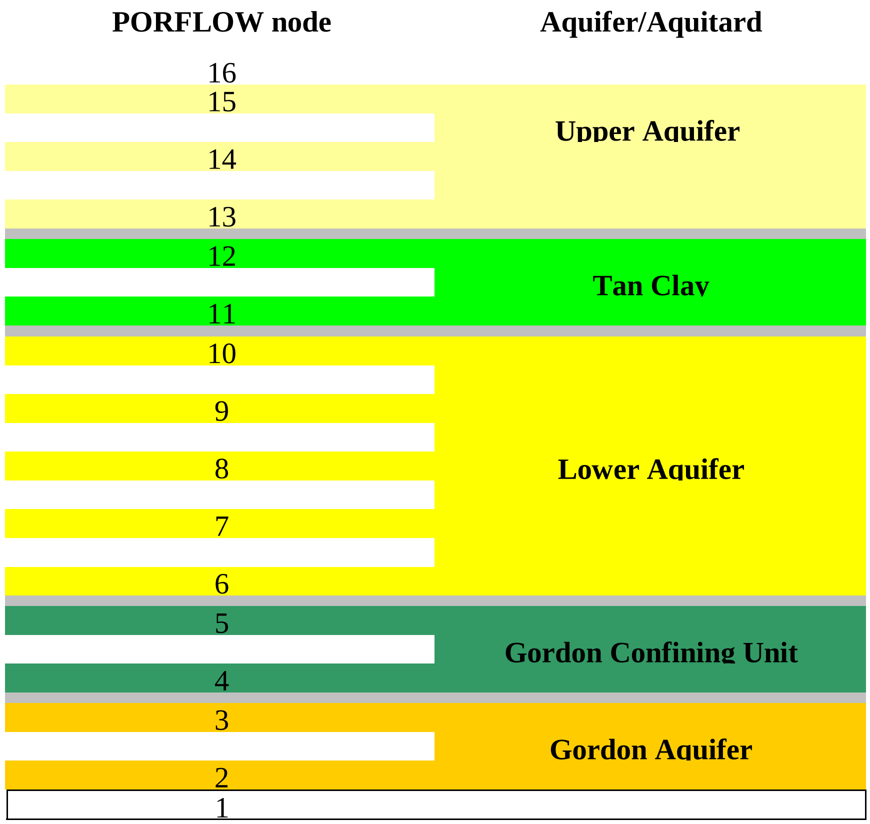
**Table 4-11. HELP Model Input Data (0 Years) (Phifer, 2003)** (continued):

| | Layer Type | Sat. Hyd. Conductivity( (cm/sec) | Drainage Length (ft) | Drain Slope (%) | Leachate Recirc. (% | Recirc. To Layer (#) | Subsurface Inflow (in/yr) |
|---|---|---|---|---|---|---|---|
| 1 | 1 | 1.00E-03 | - | - | - | - | - |
| 2 | 1 | 1.00E-04 | - | - | - | - | - |
| 3 | 1 | 3.97E-04 | - | - | - | - | - |
| 4 | 1 | 1.00E-04 | - | - | - | - | - |
| 5 | 2 | 1.00E-01 | 450 | 3 | - | - | - |
| 6 | 3 | 5.00E-09 | - | - | - | - | - |
| 7 | 1 | 1.00E-04 | - | - | - | - | - |
| 8 | 2 | 1.00E-01 | 150 | 11.4 | - | - | - |
| 9 | 3 | 5.00E-09 | - | - | - | - | - |

| | Layer Type | Geomembrane Pinhole Density (#/acre) | Geomembrane Instal. Defects (#/acre) | Geomembrane Placement Quality | Geotextile Transmissivity (cm²/sec) |
|---|---|---|---|---|---|
| 1 | 1 | - | - | - | - |
| 2 | 1 | - | - | - | - |
| 3 | 1 | - | - | - | - |
| 4 | 1 | - | - | - | - |
| 5 | 2 | - | - | - | - |
| 6 | 3 | - | - | - | - |
| 7 | 1 | - | - | - | - |
| 8 | 2 | - | - | - | - |
| 9 | 3 | - | - | - | - |

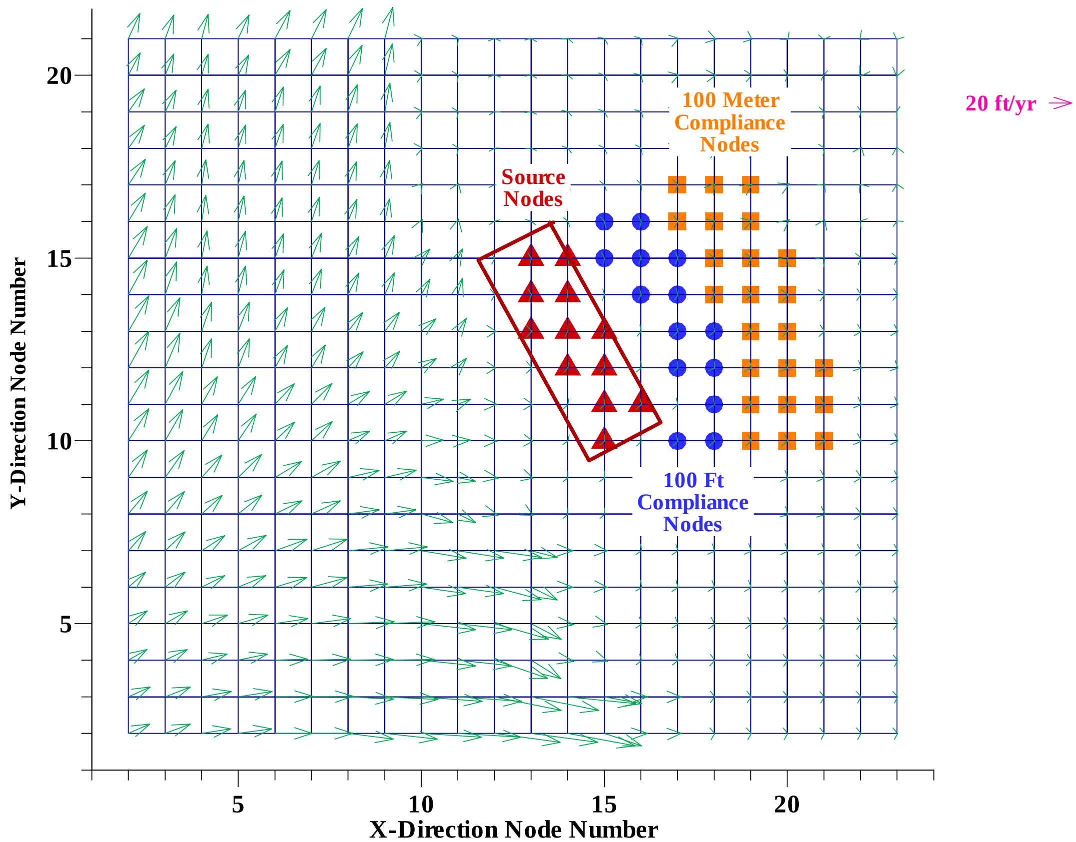
Once flow of contaminants, as a function of time, has been established through the vadose zone, the PORFLOW model simulates the transport in the groundwater to the compliance point located 100 meters away from the disposal facility. Aquifer transport simulations (Cook et al., 2005) are based on a groundwater model encompassing the General Separations Area (GSA) (Flach, 2004; Flach and Harris, 1999). The Savannah River Site F-, E-, H-, S- and Z-Areas are included in the GSA. Saltstone disposal units reside in Z-Area. The model domain is bounded by Fourmile Branch to the south, Upper Three Runs to the north, F-area to the west, and McQueen Branch to the east as shown in Figure 3-1. Vertically the model extends from the ground surface to the top of the Crouch Branch confining unit. The major hydrostratigraphic units from top to bottom are the Upper Three Runs or water table aquifer, Gordon confining unit, and Gordon aquifer. The Upper Three Runs aquifer contains a “upper” and “lower” aquifer zones separated by the “tan clay confining zone”. Figure 4-3 presents the vertical stratigraphy used in the model.

**Figure 4-3. Schematic of the Aquifer/Aquitard System Model (Cook et al., 2005)**



The location of the source and 100 meter compliance nodes for input into the PORFLOW model are shown in Figure 4-4. Groundwater flow is for the upper aquifer.

**Figure 4-4. Locations of Source Nodes and Compliance Nodes (Cook et al., 2005)**




The distribution coefficients ($K_d$) used for the modeling runs and their origin are presented in Table 4-12.

**Table 4-12. Modeling Distribution Coefficients ($K_d$) (Cook et al., 2005)**

| $K_d$ Values and References used in the Vault 4 Special Analysis | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Element | Soil $K_d$ (mL/g) | ref | Gravel $K_d$ (mL/g) | ref | Clay $K_d$ (mL/g) | ref | Saltstone Grout and Vault $K_d$ (mL/g) | ref |
| $NO_3$ | 0 | | 0 | | 0 | | 0 | |
| H | 0 | a | 0 | a | 0 | b | 0 | c |
| C | 2 | d | 2 | d | 1 | e | 5000 | c |
| K | 3 | f | | | 5 | f | 2 | f |
| Co | 8 | f | | | 96 | f | 100 | f |
| Ni | 400 | e | 400 | e | 650 | e | 100 | c |
| Se | 36 | f | 5 | g | 76 | f | 0.1 | c |
| Kr | 0 | f | | | 0 | f | 0 | f |
| Sr | 10 | j | 10 | j | 110 | e | 1 | c |
| Zr | 600 | e | 600 | e | 3300 | e | 5000 | c |
| Nb | 160 | e | 160 | e | 900 | e | 500 | c |
| Tc | 0.1 | f | 0.1 | | 0.1 | f | 1000 | c |
| Sn | 130 | e | 130 | e | 670 | e | 1000 | c |
| I | 0.6 | h | 0.6 | h | 1 | e | 2 | c |
| Cs | 330 | i | 330 | i | 1900 | e | 20 | c |
| Eu | 1900 | f | | | 8400 | f | 5000 | f |
| Pb | 270 | e | 270 | e | 550 | e | 500 | c |
| Bi | 450 | f | 450 | f | 12000 | f | 5000 | f |
| Po | 150 | e | 150 | e | 3000 | e | 500 | k |
| Rn | 0 | f | | | 0 | f | 0 | f |
| Ra | 500 | e | 500 | e | 9100 | e | 50 | c |
| Ac | 450 | e | 450 | e | 2400 | e | 5000 | l |
| Th | 3200 | e | 3200 | e | 5800 | e | 5000 | c |
| Pa | 550 | e | 550 | e | 2700 | e | 5000 | c |
| U | 800 | m | 800 | m | 1600 | e | 2000 | c |
| Np | 5 | e | 5 | e | 55 | e | 5000 | c |
| Pu | 370 | f | | | 6500 | f | 5000 | f |
| Pu_56 | 15 | f | | | 50 | f | 5000 | f |
| Am | 1900 | e | 1900 | e | 8400 | e | 5000 | c |
| Cm | 4000 | e | 4000 | e | 6000 | e | 5000 | c |
| Cf | 510 | a | 510 | a | 8400 | l | 5000 | l |

| | |
|---|---|
| a | NCRP, 1996, Table 4-1, page 44 |
| b | Used value for “soil” |
| c | Bradbury and Sarott, 1995, Table 4, page 42, Region II Reducing |
| d | McIntyre, 1988 |
| e | Sheppard and Thibault, 1990, Table 1, page 472 |
| f | Kaplan, 2004, Table 5, page 15 |
| g | Kaplan et al., 1998, Table 6, page 9 for Se |
| h | Hoeffner, 1984a, Table 2, page 5 for I |
| i | Hoeffner, 1984b, Table I, page 27 for Cs |
| j | Hoeffner, 1985, Figure 4, page 30 for Sr |
| k | Assumed to be the same as for Pb |
| l | Assumed to be the same as for Am |
| m | Serkiz and Johnson, 1994, Figure 4-12, page 69 |

The following information in Table 4-13 from the HELP modeling is also needed for the PORFLOW evaluation.

**Table 4-13. Inputs for PORFLOW Vadose Zone Modeling (Phifer, 2003)**

| Year | Infiltration through Upper GCL (in/yr) | Lower Drainage Layer Saturated Hydraulic Conductivity (cm/s) |
|---|---|---|
| 0 | 0.36 | 1.00E-01 |
| 100 | 0.41 | 1.00E-01 |
| 300 | 3.05 | 9.98E-02 |
| 550 | 7.90 | 9.89E-02 |
| 1,000 | 12.04 | 9.61E-02 |
| 1,800 | 13.76 | 8.96E-02 |
| 3,400 | 14.03 | 7.56E-02 |
| 5,600 | 14.08 | 5.62E-02 |
| 10,000 | 14.09 | 1.74E-02 |

### 4.1.5 Groundwater Pathway Code Results

The calculated maximum groundwater concentrations at any time between 100 and 10,000 years for a one curie inventory of each radionuclide in Vault 4 was entered into the LADTAP XL code in order to calculate doses via all pathways (resident farmer) as indicated in Figure 4-1. Table 4-14 presents the calculated mrem/Ci on an annual basis for those radionuclides for which an inventory limit of less than 1.0E+20 Ci is necessary assuming a performance objective of 25 mrem/yr to the whole body. The table includes the dose per radionuclide for each of the modeled pathways and assumes that the maximum concentration for each radionuclide occurs concurrently. The sensitivity for vault performance after 10,000 years is presented in section 8.4.

### Table 4-14. Maximally Exposed Individual Doses for All Groundwater Pathways

| Nuclide | Fish Consumption | Water Consumption | Shoreline | Swimming | Boating | Vegetable Consumption | Milk Consumption | Meat Consumption | Total |
|---|---|---|---|---|---|---|---|---|---|
| | Individual Dose in mrem/year/Ci Inventory | | | | | | | | |
| H-3 | 1.2E-14 | 5.1E-13 | | 2.2E-16 | | 2.2E-13 | 1.6E-13 | 5.6E-14 | 9.6E-13 |
| C-14 | 2.2E-20 | 1.8E-22 | | | | 5.7E-20 | 2.5E-20 | 2.2E-20 | 1.3E-19 |
| Al-26 | 1.5E-11 | 5.9E-11 | 8.2E-10 | 2.0E-13 | 8.4E-32 | 1.8E-10 | 1.5E-12 | 1.3E-12 | 1.1E-09 |
| Cl-36 | 1.9E-21 | 1.5E-21 | 1.7E-27 | 7.2E-32 | 8.4E-32 | 1.7E-18 | 1.2E-18 | 1.3E-18 | 4.2E-18 |
| K-40 | 8.6E-04 | 3.3E-05 | 1.7E-05 | 4.3E-09 | 5.0E-09 | 6.6E-04 | 1.7E-04 | 1.7E-04 | 1.9E-03 |
| Ni-59 | 4.5E-19 | 1.7E-19 | 2.6E-20 | 5.9E-25 | 6.9E-25 | 6.9E-19 | 1.9E-19 | 5.2E-20 | 1.6E-18 |
| Se-79 | 4.9E-04 | 1.1E-04 | | | | 8.5E-03 | 1.4E-02 | 1.6E-03 | 2.5E-02 |
| Rb-87 | 3.2E-12 | 6.2E-14 | | | | 6.3E-13 | 7.2E-13 | 2.6E-13 | 4.9E-12 |
| Sr-90 | 2.5E-17 | 3.2E-17 | | | | 1.1E-16 | 3.8E-18 | 9.8E-19 | 1.7E-16 |
| Nb-93m | 1.7E-04 | 2.1E-07 | 1.3E-09 | 9.8E-14 | 1.2E-13 | 6.8E-07 | 7.2E-08 | 2.8E-06 | 1.7E-04 |
| Nb-94 | 3.4E-17 | 4.4E-20 | 9.9E-19 | 2.1E-22 | 2.5E-22 | 1.5E-19 | 1.6E-20 | 6.0E-19 | 3.6E-17 |
| Mo-93 | 7.0E-07 | 2.7E-06 | 8.7E-07 | 2.8E-11 | 3.3E-11 | 2.6E-05 | 7.4E-06 | 2.8E-06 | 4.0E-05 |
| Tc-99 | 7.4E-19 | 1.9E-18 | 6.7E-23 | 1.3E-26 | 1.5E-26 | 3.2E-17 | 3.0E-17 | 1.7E-16 | 2.3E-16 |
| Pd-107 | 2.4E-17 | 9.3E-17 | | | | 8.0E-16 | 1.5E-16 | 1.1E-17 | 1.1E-15 |
| Sn-126 | 7.8E-19 | 9.9E-21 | 2.6E-21 | 4.7E-25 | 5.5E-25 | 3.0E-20 | 3.2E-21 | 3.5E-20 | 8.6E-19 |
| I-129 | 3.7E-04 | 9.4E-04 | 5.4E-06 | 5.0E-10 | 5.9E-10 | 3.8E-03 | 9.3E-04 | 1.6E-04 | 6.2E-03 |
| Ra-226 | 1.1E-16 | 8.4E-17 | 4.2E-20 | 8.1E-24 | 9.6E-24 | 4.3E-16 | 1.7E-17 | 6.0E-18 | 6.5E-16 |
| Np-237 | 1.7E-19 | 6.5E-19 | 3.9E-22 | 6.2E-26 | 7.3E-26 | 2.0E-18 | 4.2E-22 | 5.7E-21 | 2.8E-18 |

## 4.2 Air Pathways Analysis

### 4.2.1 Air Pathways Methodology

Air pathways are pathways by which an individual may receive a dose due to gaseous radionuclides diffusing from the vault to the soil surface and into the environment. Dose may come from direct plume shine, inhalation, and ingestion of vegetables, meat, and milk exposed to airborne radioactivity as presented in Figure 4-1.

### 4.2.2 Radionuclide Screening

A screening analysis was conducted to produce a list of radionuclides requiring a detailed analysis for the air pathway. The screening analysis (Crapse and Cook, 2004) used a methodology developed by the NCRP (NCRP, 1996) to determine the list of radionuclide of interest for the air pathways. The 826 radionuclides originally considered potentially significant were reduced to 9 to be evaluated in detail: C-14, Cl-36, H-3, I-129, Sb-125, Sb-126, Se-79, Sn-121m, and Sn-126.

### 4.2.3 Computer Codes/Hand-Analysis

Two computer codes and a hand-analysis were used to calculate dose to individuals from air pathways from Saltstone Vault 4 which will be transitioned to the entire facility in Section 4.3.2. The computer codes are the PORFLOW numerical model (ACRI, 2002) and the EPA computer code CAP88 (Chaki, 2002).

The PC-based PORFLOW Version 5.97.0 was used to conduct a series of simulations to evaluate transient radionuclide transport through the soil cover above Saltstone Vault 4 and to determine the gaseous radionuclide flux at the land surface over time. PORFLOW is developed and marketed by Analytic and Computational Research, Inc and has been widely used at SRS and in the DOE complex.

The EPA computer code Clean Air Act Assessment Package-1988 (CAP88) Version 1.0 is used to determine airborne transport and doses for radionuclides. The code calculates dose due to air pathways to the maximally exposed individual (MEI) based on input parameters specified by the user. CAP88 can account for such parameters as radioactive decay, plume transit, and meteorological data. Doses are calculated for direct plume shine, inhalation, and ingestion of vegetables, meat, and milk exposed to airborne radioactivity

The base data including dose conversion factors built into the code, meteorological data (Weber, 2002), and consumption data (Hamby, 1991**)** are based on the best available SRS data. SRS has received approval from EPA to adjust the humidity value in CAP88 from

8 gm/$m^3$ to 11 gm/$m^3$ (USDOE, 1991 and USEPA, 1991). The CAP88 code has been verified by the EPA (Chaki, 2002).

Selenium-79 is not contained within the CAP88 dose factor library; therefore a hand calculation using the methods in Simpkins (1999) was used to estimate the dose at 100 m. The hand calculation is based on formulas found in NRC Regulatory Guides (1977a, 1977b). The calculated dose is based on direct plume shine, inhalation, and ingestion of vegetables, meat, and milk exposed to airborne radioactivity, though there is no shine component from Se-79. The meteorological and consumption data come from the same sources as for CAP88, but the dose conversion factors come from the USDOE (1988b, 1988c).

### 4.2.4 Analysis Inputs/Assumptions

#### 4.2.4.1 Diffusion Flux Rates

PORFLOW was used to evaluate transient radionuclide transport through the soil cover above Saltstone Vault 4, as presented in Figures 3-4 and 3-5 and Table 3-1, in order to determine the gaseous radionuclide flux at the land surface.

The evaluation (Cook et al., 2005) assumed the following forms for the gaseous state of the radionuclides:

- C-14 exists as part of the $CO_2$ molecule
- Cl-36, H-3, and I-129 exist as diatomic gasses
- Sb-125, Se-79, Sn-121m, and Sn-126 exist as monatomic gasses.

The PORFLOW code used only the diffusive and net decay terms and disabled the advection term. Flow field was assumed to be isobaric and isothermal. The boundary conditions imposed on the model were

- No-flux specified for all radionuclides along sides and bottom
- Radionuclide concentrations set to 0 at land surface.

Values for total porosity and long-term residual saturation of the closure cap materials were based on the investigation summarized in Phifer and Nelson (2003) and are given in Table 4-15.

**Table 4-15. Porosity, Residual Saturation and Air-Filled Porosity Values (Phifer and Nelson, 2003)**

| Layer Material | Representative Porosity | Long-term Residual Saturation | Air-filled Porosity |
|---|---|---|---|
| Erosion Barrier | 0.07 | 0.83 | 1.19E-02 |
| Upper Backfill | 0.38 | 0.63 | 1.39E-01 |
| Upper Drainage | 0.38 | 0.58 | 1.58E-01 |
| Lower Backfill | 0.37 | 0.72 | 1.04E-01 |
| Lower Drainage | 0.31 | 0.5 | 1.60E-01 |
| Concrete | 0.18 | 0.99 | 2.00E-03 |
| Saltstone | 0.42 | 0.99 | 4.00E-03 |

Molecular diffusion coefficients for each radionuclide were calculated based on the effective open air diffusion coefficient of radon, as reported in Nielson et al. (1984). The calculated coefficients are given in Table 4-16.

**Table 4-16. Effective Air-Diffusion Coefficients for Each Radionuclide/Compound, by Material (Nielson et al., 1984)**

| Radionuclide | Saltstone and Concrete ($m^2$/yr) | Lower Drainage ($m^2$/yr) | Lower Backfill ($m^2$/yr) | Upper Drainage and Upper Backfill ($m^2$/yr) | Erosion Barrier ($m^2$/yr) |
|---|---|---|---|---|---|
| $^{14}CO_2$ | 4.86E-01 | 1.39E+01 | 6.24E+00 | 1.04E+01 | 1.73E+00 |
| $^{36}Cl_2$ | 3.88E-01 | 1.11E+01 | 4.99E+00 | 8.31E+00 | 1.39E+00 |
| $^{3}H_2$ | 1.34E+00 | 3.84E+01 | 1.73E+01 | 2.88E+01 | 4.80E+00 |
| $^{129}I_2$ | 2.05E-01 | 5.85E+00 | 2.63E+00 | 4.39E+00 | 7.32E-01 |
| $^{125}Sb$ | 2.95E-01 | 8.41E+00 | 3.78E+00 | 6.30E+00 | 1.05E+00 |
| $^{126}Sb$ | 2.93E-01 | 8.38E+00 | 3.77E+00 | 6.28E+00 | 1.05E+00 |
| $^{79}Se$ | 3.70E-01 | 1.06E+01 | 4.76E+00 | 7.93E+00 | 1.32E+00 |
| $^{121m}Sn$ | 2.99E-01 | 8.55E+00 | 3.85E+00 | 6.41E+00 | 1.07E+00 |
| $^{126}Sn$ | 2.93E-01 | 8.38E+00 | 3.77E+00 | 6.28E+00 | 1.05E+00 |

#### 4.2.4.2 Dose Release Factors

The dose release factors (DRF) to the maximally exposed individuals (MEI) were determined at 100 m (Simpkins, 2004b). The release was assumed to be from ground level and to occur over one year. CAP88 was used to calculate the DRF due to a point source, except for Se-79 which used hand calculations. CAP88 has the ability to handle area source, but the model is not deemed to be appropriate close to the source (Moore et al., 1979). Instead, hand calculations were performed for a point versus area source for average meteorological conditions (D stability, 4.5 m/s wind speed) (USDOE, 1997), and the ratio used to determine the CAP88 point source DRF decrease due to an area release. Using the Pasquill Briggs Diffusion coefficient (Moore et al., 1979), the vertical diffusion coefficient at 100 m was calculated to be 5.6 m (Simpkins, 2004b). The sector-average relative air concentration for a point source was estimated to be 8.1E-04 s/m$^3$ using a Gaussian plume equation (USNRC, 1977b). The sector-average relative air concentration for an area source the size of Vault 4 was estimated to be 1.0E-04 s/m$^3$ using the formula by Napier, (2002) a factor of 8 below the point source estimate. The CAP88 point source values were reduced by a factor of 5 to determine the Vault 4 DRFs. The reduction factor was reduced from 8 to 5 to account for the fact that actual meteorological data was not used.

#### 4.2.4.3 MEI Dose

The dose to the MEI from 1 Ci of radionuclide in the vault was determined by multiplying the dose release factor at 100 m by the maximum diffusion flux rate calculated after 100 yrs. Note: Due to the short half-life of Sb-125 (2.77 yrs), its maximum dose to the MEI occurs at the site boundary during the 100-yr period of institutional control.

### 4.2.5 Analysis Results

The radionuclide Sb-126 is a daughter product of, and in equilibrium with, Sn-126. Therefore, the dose for Sb-126 is not calculated individually, but combined into the dose for Sn-126 (Cook et al., 2005). The SDF Vault 4 dose results due to air pathways are presented in Table 4-17 (Simpkins, 2004b).

**Table 4-17. SDF Vault 4 Dose Results (Cook et al., 2005)**

| Radionuclide | Dose to MEI at 100 m from 1 Ci in the vault (mrem/yr) |
|---|---|
| C-14 | 2.28E-07 |
| Cl-36 | 6.81E-19 |
| H-3 | 1.83E-11 |
| I-129 | 5.90E-14 |
| Sb-125* | 3.36E-47 |
| Se-79 | 2.07E-06 |
| Sn-121m | 4.32E-67 |
| Sn-126 | 1.55E-62 |

*Sb-125 dose is at the site boundary due to its 2.77-yr half-life

## 4.3 All-Pathways Analysis

In this PODD, exposures from all-pathways are calculated using the peak groundwater doses calculated from the groundwater pathways analysis (Section 4.1) and the peak air doses calculated in the air pathways analysis (Section 4.2).

### 4.3.1 Methodology

Figure 4-1 depicts the all-pathways modeled for this demonstration analysis.

The groundwater pathways doses calculated in Section 4.1 include those of the resident farmer who uses the contaminated groundwater at the 100-meter well as a source of 1) drinking water, 2) pond water (in which fish are raised and recreational activities occur), and 3) irrigation water used for raising vegetables, meat, and milk. The groundwater pathways doses are calculated for a unit curie inventory for each radionuclide.

The air pathway doses calculated in Section 4.2 include not only direct radiation and inhalation from the airborne plume but also doses from consumption of vegetables, meat, and milk contaminated from the airborne plume. The air pathway dose is also calculated for a unit curie inventory of each radionuclide.

The all-pathways dose from the groundwater pathway from Table 4-14 and the all-pathways dose from the air pathway from Table 4-17 are summed to obtain the total all-pathways dose. The total all-pathways dose per curie is ratioed with the all-pathways

performance objective from 10 CFR 61.41 of 25 mrem/year to the whole body to obtain the all-pathways limit for each radionuclide.

### 4.3.2 All-Pathways Performance Objective Demonstration

Table 4-18 presents the all-pathways limits for the 10,000-year time frame for Vault 4.

**Table 4-18. All-Pathways Inventory Limits for SDF Vault 4 (Cook et al., 2005)**

| Radionuclide | 10,000-Year Inventory Limit (Ci/Vault 4) |
|---|---|
| H-3 | 1.30E+12 |
| C-14 | 1.10E+08 |
| Al-26 | 2.31E+10 |
| Cl-36 | 5.15E+18 |
| K-40 | 1.31E+04 |
| Ni-59 | 1.58E+19 |
| Se-79 | 1.02E+03 |
| Rb-87 | 5.12E+09 |
| Sr-90 | 1.42E+17 |
| Nb-93m | 1.46E+05 |
| Nb-94 | 6.98E+17 |
| Mo-93 | 6.17E+05 |
| Tc-99 | 1.07E+17 |
| Pd-107 | 1.84E+17 |
| Sn-126 | 2.92E+19 |
| I-129 | 4.03E+03 |
| Ra-226 | 3.84E+16 |
| Np-237 | 8.93E+18 |

The new dose results to demonstrate compliance are obtained by conservatively assuming that the entire inventory of salt waste radioactivity (including existing Saltstone inventory in Vaults 1 and 4 and projected inventory for all future vaults) is contained in Vault 4 and the maximum concentration of each radionuclide in the two media (groundwater and atmospheric) occurs at the same time for each radionuclide. The projected inventory for all of the SDF is presented in Table 3-2. The projected inventory is compared to the Vault 4 limits from Cook et al., (2005) and based on the all-pathways performance objective of 25 mrem/yr to the whole body and the sum-of-fractions of the inventory limits indicates a total whole body dose of 2.3 mrem/yr as presented in Table 4-19.

**Table 4-19. Evaluation of All-Pathways Doses**

| Radionuclide | 10,000-Year Disposal Limit (Ci/Vault 4)* | Total Saltstone Inventory (Ci) | Fraction of 10,000-Year Disposal Limit | Dose (mrem/yr) |
|---|---|---|---|---|
| H-3 | 1.30E+12 | 9.43E+03 | 7.25E-09 | 1.81E-07 |
| C-14 | 1.10E+08 | 5.20E+02 | 4.72E-06 | 1.18E-04 |
| Al-26 | 2.31E+10 | 2.35E+01 | 1.02E-09 | 2.54E-08 |
| Ni-59 | 1.58E+19 | 2.85E+00 | 1.81E-19 | 4.52E-18 |
| Se-79 | 1.02E+03 | 8.94E+01 | 8.77E-02 | 2.19E+00 |
| Sr-90 | 1.42E+17 | 7.43E+03 | 5.23E-14 | 1.31E-12 |
| Nb-94 | 6.98E+17 | 4.22E-03 | 6.05E-21 | 1.51E-19 |
| Tc-99 | 1.07E+17 | 3.31E+04 | 3.10E-13 | 7.74E-12 |
| Sn-126 | 2.92E+19 | 4.51E+02 | 1.54E-17 | 3.86E-16 |
| I-129 | 4.03E+03 | 1.80E+01 | 4.46E-03 | 1.12E-01 |
| Ra-226 | 3.84E+16 | 1.30E+01 | 3.39E-16 | 8.46E-15 |
| Np-237 | 8.93E+18 | 2.12E+00 | 2.37E-19 | 5.93E-18 |
| | | **Totals** | **9.21E-02** | **2.30E+00** |

* Vault 4 inventory limits from Table 6-1 of Cook et al., (2005) based upon all-pathways dose limit of 25 mrem/yr to the whole body

The whole body dose is a result of two principle dose contributors: Se-79 and I-129. The whole body dose from these two radionuclides is principally a result of the ingestion pathway. USEPA (1988) values for ingestion dose conversion factors are utilized to determine doses to other organs by determining the ratio of the organ dose conversion factors to the whole body factor and multiplying by the known whole body dose. The final results (Table 4-20) indicate that for salt waste disposal at the SDF the all-pathways doses are 2.3 mrem/yr whole body, 4.6 mrem/yr to the thyroid and 5.3 mrem/yr to any other organ. This is compared to 10 CFR 61.41 performance objectives of 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ and demonstrates that the 10 CFR 61.41 performance objectives are met by the SDF.

**Table 4-20. 10 CFR 61.41 Demonstration Results**

| | **10 CFR 61.41 Limit (mrem/year)** | **SDF Calculated Dose (mrem/year)** |
|---|---|---|
| **Whole Body** | 25 | 2.3 |
| **Thyroid** | 75 | 4.6 |
| **Any Other Organ** | 25 | 5.3 |

# 5.0 10 CFR 61.42 Compliance (Inadvertent Intruder Analysis)

In estimating doses to inadvertent intruders after the period of active institutional control (i.e., at any time beyond 100 years after closure of the disposal facility), it is assumed that such individuals could establish a permanent homestead on the site. Furthermore, it is assumed that an intruder has no *a priori* knowledge of waste disposal activities at the site. For direct intrusion into SDF vaults after loss of active institutional control, exposures are assumed to occur according to one of three scenarios: agriculture, resident and post-drilling scenarios. For compliance of SDF vaults with the 10 CFR 61.42 intruder performance objective, all of the vaults are bounded by the inventory in Vault 4. This is conservative because Cs-137 is the dominant radionuclide for intruder dose and the DDA stream, which has the highest Cs-137 concentration, will be located primarily in Vault 4.

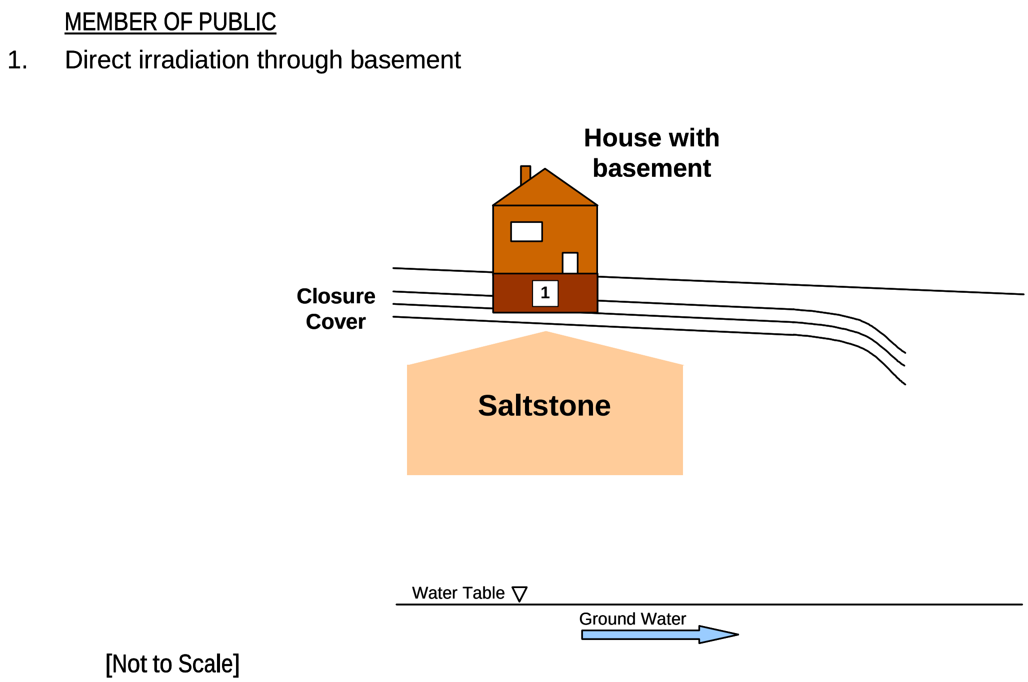
Evaluation of intruder scenarios can include either chronic exposure or single acute exposure. Following evaluation of intruder scenarios (Kennedy and Peloquin, 1988), chronic scenarios were determined to be more limiting because of the longer exposure times. Therefore based on the evaluation, only chronic exposures have been evaluated for the SDF.

Doses to a hypothetical inadvertent intruder are estimated based on assumptions about credible exposure scenarios at different times after disposal and their associated exposure pathways. The scenarios for inadvertent intrusion at different times are based on an assumed design and performance of the cover system above a disposal vault. (Phifer and Nelson, 2003)

## 5.1 Definition of Intruder Scenarios

The resident scenario is a credible occurrence at any time after institutional control is relinquished. At 100 years after disposal, all engineered barriers above the waste are assumed to be intact. An assumption that the barriers to excavation will not degrade by a significant amount during the 100-year period of institutional control is reasonable when surveillance and maintenance of the cover system presumably will be performed during that time. An inadvertent intruder then is assumed to excavate to the depth of 3 m, a typical maximum depth of an excavation in digging a foundation for a home. Over time erosion will lower the ground elevation, until the erosion barrier becomes exposed. Thus the amount of shielding will decrease and radioactive decay of long-lived radionuclides will produce increasing quantities of daughter products. The intruder analysis was performed in ten-year steps from year 100 to year 10,000 to find the maximum contribution from each radionuclide. Figure 5-1 presents the resident intruder pathway.

**Figure 5-1. Resident Intruder Pathway**




It is assumed in this analysis that the overall disposal system will perform as an effective barrier to inadvertent intrusion over 10,000 years (Cook et al., 2005). The erosion barrier is constructed of material sized to remain in place during a rainfall event with a 10,000-year recurrence interval (Phifer and Nelson, 2003, Appendix K). The cover system therefore provides a distance of greater than 10 feet from the top of the erosion barrier to the Saltstone waste form over the 10,000-year time frame and therefore the 10 foot basement construction does not contact the waste material and therefore the agricultural scenario is not possible (see Figure 3-5 for closure cap dimensions). The agricultural scenario then becomes the resident scenario, where a home is constructed with a basement that does not bring up waste material.

The reinforced concrete vault roof and the Saltstone waste form present a barrier that would provide reasonable assurance that drilling activities would be discouraged and therefore drilling is not considered a probable scenario (Cook et al., 2005). The persistent, thick cover system provides protection from physical weathering. The concrete and saltstone grout will undergo chemical degradation over time, which will slowly alter the nature of the cementitious materials. Initially and for many years afterward, the roof and saltstone grout will present a dramatically more difficult media through which to drill. The rational response would be to move to a nearby location where his equipment could penetrate to the desired depth. In later times, the nature of the altered concrete and saltstone grout will still present a sharp contrast to the native sand and clays, which

should provide enough information to an inadvertent intruder so that he would conclude he is not dealing with naturally occurring materials. Therefore the post-drilling scenario was deemed to not be credible (Cook et al., 2005).

Because of the depth of the waste, the agriculture scenario was not included as part of the intruder analysis, it was evaluated as a sensitivity calculation in Section 8. Likewise, the post-drilling scenario was not included as part of the intruder analysis but was evaluated as a sensitivity calculation in Section 8.

## 5.2 Intruder Analysis Radionuclide Screening

The radionuclides to be evaluated for the resident scenario are those deemed important from a screening analysis (Cook and Wilhite, 2004). The screening analysis used a methodology developed by the NCRP which began with 826 radionuclides considered potentially significant (NCRP, 1996). The result of the screening analysis was 132 radionuclides that were evaluated in detail for the resident scenario. The radionuclides are presented in Appendix B with half-life data are taken from Tuli (2000). Branching fractions are taken from International Commission on Radiological Protection (ICRP) 38 and are listed for radionuclides that have radioactive daughters (ICRP, 1983).

## 5.3 Intruder Analysis Computational Code

The intruder analysis is performed by an automated inadvertent intruder analysis code developed by Savannah River National Laboratory (SRNL) (Koffman, 2004). The updated intruder analysis model includes the Bateman equation and therefore estimates disposal limits for all progeny in decay chains. The code calculates intruder doses based upon dose conversion factors and specific input parameters that can be input by the user. The code can account for such parameters as radionuclide decay, decay product in-growth, and shield thickness degradation while determining intruder doses at specified intervals over time. The intruder analysis code was set to perform an analysis in ten-year steps from year 100 to year 10,000 to find the maximum contribution from each radionuclide.

The analysis code methodology has been checked via comparison to previously approved spreadsheet methodology results in Appendix B of Koffman (2004). By this comparison, the code methodology and the use of the base data (Lee, 2004) are verified.

## 5.4 Intruder Analysis Computational Code Input/Assumptions

### 5.4.1 Intruder Input - Internal and External Dose Conversion Factors (DCFs)

Radionuclide Dose Conversion Factors (DCFs) used to calculate the dose equivalents are taken from Federal Guidance Reports (FGRs) developed by the EPA (USEPA, 1988 and

1993). Shielding dose coefficients developed by Kocher (2004) are used to evaluate dose equivalents for external exposure to various thicknesses of contaminated soil. These values are listed in Appendix B

#### 5.4.1.1 Internal DCFs

Ingestion and inhalation DCFs are taken from FGR 11 (USEPA, 1988). Internal DCFs in FGR 11 represent the 50-year committed effective dose equivalent per unit of activity, reported in SI units (Sv/Bq). These factors are included in the intruder analysis code (Koffman, 2004).

#### 5.4.1.2 External DCFs

External DCFs for uniformly distributed contamination at an infinite depth with no shielding and at 15 cm are taken from FGR 12 (USEPA, 1993). External DCFs in FGR 12 represent the 50-year committed effective dose equivalent per unit of activity of soil contaminated at various depths. These factors are included in the intruder analysis code (Koffman, 2004).

### 5.4.2 Intruder Analysis Computational Code Input-External Pathway Shielding Dose Coefficients

For soil contaminated at various depths shielded by a layer of clean soil, external dose coefficients for absorbed dose at 1 m from a mono-energetic source were estimated by (Kocher and Sjoreen, 1985). A comprehensive list of absorbed dose rates at finite thicknesses were provided by Kocher (2004) based on the work presented in (Kocher and Sjoreen, 1985). These data were used to estimate shielding effective dose coefficients at various shielding thicknesses with contamination at finite depths.

Coefficients for shielding depths of 5 and 100 cm are used in the intruder analysis model (Koffman, 2004) to estimate dose coefficients at various depths for the residential exposure scenario. Appendix C lists the external pathway shielding dose coefficients.

### 5.4.3 Intruder Analysis Computational Code Input – Physical Parameters

Table 5-1 presents the parameters necessary to calculate the intruder doses for the SDF Vault 4.

**Table 5-1. Intruder Parameters for Vault 4 (Cook et al., 2005)**

| Parameter | Value | Reference |
|---|---|---|
| Resident Geometry Factor | 0.6 | Cook et al., 2002 |
| Post-Drilling Geometry Factor | 1 | Cook et al., 2002 |
| Waste Volume ($m^3$) | 78800 | Cook et al., 2002 |
| Resident Analysis Start Time (yr) | 100 | |
| Post-Drilling Analysis Start Time (yr) | 1000 | |
| Resident Shielding Thickness (cm) | 100 | |
| Density of soil ($kg/m^3$) | 1400 | Baes and Sharp 1983 |
| Dilution factor for mixing of waste with garden soil | | |
| agriculture scenario | 0.2 | Napier et al. 1984 |
| post drilling scenario | 0.02 | Cook et al., 2002 |
| Air mass loading of soil particulates ($kg/m^3$) | | |
| working in garden | 1.0E-07 | Cook et al., 2002 |
| residing in home | 1.0E-08 | Cook et al. 2002 |
| Consumption of contaminated drinking water (L/yr) | 730 | Lee, 2004 |
| Consumption of contaminated vegetables (kg/yr) | 90 | Lee, 2004 |
| Air intake (breathing rate) ($m^3$/yr) | 8000 | USNRC 1977a |
| Consumption of contaminated soil (kg/yr) | 0.037 | USEPA 1989 |
| Exposure time as fraction of year (/yr) | | |
| Working in garden | 0.01 | Oztunali et al. 1981 |
| Residing in home | 0.50 | Oztunali et al. 1981 |
| Shielding factor of home for external exposure during indoor residence | 0.7 | USNRC 1977a |
| Dose Limit (rem/yr) | 0.1 | USDOE 1999 |

**Transient Layer Model (Surface to Top of Waste)** (Phifer and Nelson 2004)

| Layer | Thickness (m) | Description | Erosion Rate (mm/yr) | Degradation Time (yr) |
|---|---|---|---|---|
| 1 | 0.9144 | Soil cover (36") | 1.4 | 0 |
| 2 | 0.3048 | Erosion barrier (12") | 1.00E-10 | 0 |
| 3 | 2.7178 | Soil backfill (107") | 1.4 | 0 |
| 4 | 0.5080 | Concrete/Grout Min (20") | 1.4 | 1000 |

Element plant-to-soil ratios present in Table 5-2 are taken from Baes et al (1984) unless noted below. These are contaminant specific ratios of fresh weight in vegetation (μCi/kg) per dry weight in soil (μCi/kg). Values taken from Baes et al. (1984) are reported in dry weight of vegetation and are multiplied by 0.43 to get fresh weight.

**Table 5-2. Element Plant-to-Soil Ratios in Vegetables**

| Element | Soil Ratio | Element | Soil Ratio |
|---|---|---|---|
| Ac | 1.51E-04 | Np | 4.30E-03 |
| Am | 1.08E-04 | Pa | 1.08E-04 |
| At | 6.45E-02 | Pb | 3.87E-03 |
| Ba | 6.45E-03 | Pd | 1.72E-02 |
| Bi | 2.15E-03 | Po | 1.72E-04 |
| Bk | 6.60E-06 | Pu | 1.94E-05 |
| C* | 5.60E-01 | Ra | 6.45E-03 |
| Ca | 1.51E-01 | S | 6.45E-01 |
| Cd | 6.50E-02 | Sb | 1.29E-02 |
| Cf | 6.60E-06 | Sc | 4.30E-04 |
| Cl | 3.01E+01 | Se | 1.08E-02 |
| Cm | 6.45E-06 | Sm | 1.72E-03 |
| Co | 3.01E-03 | Sn | 2.58E-03 |
| Cs | 1.29E-02 | Sr | 1.08E-01 |
| Eu | 1.72E-03 | Tc | 6.45E-01 |
| Fr | 1.29E-02 | Th | 3.66E-05 |
| Gd | 1.72E-03 | Tl | 1.72E-04 |
| H** | 4.80E+00 | U | 1.72E-03 |
| I | 2.15E-02 | W | 4.30E-03 |
| K | 2.37E-01 | Y | 2.58E-03 |
| Mo | 2.20E-03 | Zr | 2.15E-04 |
| Nb | 2.15E-03 | | |

*C is based on (Sheppard, et al., 1990)
**H obtained from (USNRC, 1977a)

## 5.5 Intruder Analysis Code Results

SDF Vault 4 inventory limits from Cook et al., (2005) were developed based on an intruder performance objective of 100 mrem/year per USDOE (2001). 500 mrem is chosen as the 10 CFR 61.42 performance objective because it is the basis for 10 CFR 61 waste classification discussed in Section 5.2 of the Final Environmental Impact Statement on 10 CFR 61 (USNRC, 1982). While the 10 CFR 61.42 performance objective evaluated is 500 mrem/year (USNRC, 1982), the inventory limits can still be used to calculate an intruder dose using a sum-of-fractions method. The results of the analysis of the resident scenario for the period 100 to 10,000 years are presented in Table 5-3. While all radionuclides from Appendix B were evaluated, Table 5-3 includes only those radionuclides with inventory limits less than 1.0E+20 Ci.

**Table 5-3. Intruder-Based Radionuclide Disposal Limits for Vault 4 Resident Scenario with Transient Calculation for 100 – 10,000 Years**

| Radionuclide | Time of Limit (Years) | Inventory Limit (Ci/Unit) |
|---|---|---|
| Na-22 | 100 | 7.80E+15 |
| Al-26 | 760 | 1.61E+02 |
| K-40 | 760 | 3.15E+03 |
| Co-60 | 100 | 5.75E+09 |
| Kr-85 | 100 | 2.73E+11 |
| Nb-94 | 760 | 1.01E+03 |
| Tc-99 | 760 | 3.66E+13 |
| Ag-108m | 760 | 5.68E+03 |
| Sn-126 | 760 | 1.17E+03 |
| Sb-125 | 100 | 1.41E+17 |
| Cs-134 | 100 | 4.12E+19 |
| Cs-137 | 100 | 5.99E+06 |
| Ba-133 | 100 | 1.21E+10 |
| Eu-152 | 100 | 6.42E+06 |
| Eu-154 | 100 | 1.15E+08 |
| Eu-155 | 100 | 1.12E+19 |
| Pb-210 | 100 | 3.94E+11 |
| Bi-207 | 100 | 3.08E+05 |
| Ra-226 | 760 | 4.21E+02 |
| Ra-228 | 100 | 3.72E+08 |
| Ac-227 | 100 | 8.78E+07 |
| Th-228 | 100 | 1.88E+19 |
| Th-229 | 760 | 8.61E+03 |
| Th-230 | 9090 | 3.29E+02 |

**Table 5-3. Intruder-Based Radionuclide Disposal Limits for Vault 4 Resident Scenario with Transient Calculation for 100 – 10,000 Years**

| Radionuclide | Time of Limit (Years) | Inventory Limit (Ci/Unit) |
|---|---|---|
| Th-232 | 760 | 1.56E+02 |
| Pa-231 | 760 | 2.15E+04 |
| U-232 | 100 | 9.00E+03 |
| U-233 | 10000 | 1.35E+04 |
| U-234 | 10000 | 4.48E+03 |
| U-235 | 10000 | 1.03E+05 |
| U-236 | 10000 | 3.17E+08 |
| U-238 | 10000 | 6.60E+04 |
| Np-237 | 10000 | 6.73E+04 |
| Pu-238 | 10000 | 1.27E+07 |
| Pu-239 | 10000 | 1.37E+10 |
| Pu-240 | 10000 | 2.96E+12 |
| Pu-241 | 10000 | 1.02E+10 |
| Pu-242 | 10000 | 4.91E+10 |
| Pu-244 | 760 | 3.65E+03 |
| Am-241 | 10000 | 3.38E+08 |
| Am-242m | 10000 | 9.83E+06 |
| Am-243 | 760 | 2.96E+05 |
| Cm-242 | 10000 | 2.51E+09 |
| Cm-243 | 100 | 7.00E+09 |
| Cm-244 | 10000 | 1.08E+15 |
| Cm-245 | 760 | 8.42E+06 |
| Cm-246 | 10000 | 8.34E+12 |
| Cm-247 | 10000 | 2.45E+04 |
| Cm-248 | 10000 | 4.64E+07 |
| Bk-249 | 760 | 4.92E+07 |
| Cf-249 | 760 | 1.27E+05 |
| Cf-250 | 10000 | 3.05E+15 |
| Cf-251 | 760 | 1.83E+06 |
| Cf-252 | 10000 | 6.31E+12 |

### 5.6 Intruder Performance Objective Demonstration

These limits for the intruder pathway are compared with limits derived for the other pathways and with the projected Vault 4 inventory in Table 3.2. For the projected Vault 4 inventory, only Cs-137 produces a significantly large fraction of the intruder limit.

For the projected Vault 4 inventory, the dose to the inadvertent intruder presented in Table 5-4 from the resident scenario, which is the only credible scenario within the 10,000-year time frame, is 21.7 mrem/year (Cook et al., 2005), which is 4% of the NRC performance objective of 500 mrem/year (USNRC, 1982).

Since the highest Cs-137 concentration per vault is from DDA material in Vault 4 and any other nuclide with a lower inventory limit, such as Sn-126, is spread out among future vaults and thus is not concentrated in any individual vault, the intruder dose for Vault 4 bounds future operations.

**Table 5-4. Evaluation of Inadvertent Intruder Doses (Cook et al., 2005)**

| Radionuclide | 10,000-Year Disposal Limit (Ci/Vault 4)* | Vault 4 Projected Inventory (Ci)** | Fraction of 10,000-Year Disposal Limit | Dose (mrem/yr) |
|---|---|---|---|---|
| Na-22 | 7.80E+15 | 2.59E+02 | 3.32E-14 | 3.32E-12 |
| Al-26 | 1.61E+02 | 1.03E+00 | 6.40E-03 | 6.40E-01 |
| Co-60 | 5.75E+09 | 4.46E+01 | 7.76E-09 | 7.76E-07 |
| Nb-94 | 1.01E+03 | 1.02E-03 | 1.01E-06 | 1.01E-04 |
| Tc-99 | 3.66E+13 | 7.16E+02 | 1.95E-11 | 1.95E-09 |
| Sn-126 | 1.17E+03 | 9.56E+00 | 8.17E-03 | 8.17E-01 |
| Sb-125 | 1.41E+17 | 2.05E+02 | 1.45E-15 | 1.45E-13 |
| Cs-134 | 4.12E+19 | 2.40E+03 | 5.83E-17 | 5.83E-15 |
| Cs-137 | 5.99E+06 | 1.20E+06 | 2.00E-01 | 2.00E+01 |
| Eu-152 | 6.42E+06 | 1.48E+00 | 2.30E-07 | 2.30E-05 |
| Eu-154 | 1.15E+08 | 8.10E+01 | 7.04E-07 | 7.04E-05 |
| Eu-155 | 1.12E+19 | 1.72E+01 | 1.54E-18 | 1.54E-16 |
| Ra-226 | 4.21E+02 | 2.44E-01 | 5.80E-04 | 5.80E-02 |
| Ra-228 | 3.72E+08 | 6.41E-06 | 1.72E-14 | 1.72E-12 |
| Ac-227 | 8.78E+07 | 1.37E-06 | 1.56E-14 | 1.56E-12 |
| Th-229 | 8.61E+03 | 2.79E-03 | 3.24E-07 | 3.24E-05 |
| Th-230 | 3.29E+02 | 1.49E-03 | 4.53E-06 | 4.53E-04 |
| Th-232 | 1.56E+02 | 6.41E-06 | 4.11E-08 | 4.11E-06 |
| Pa-231 | 2.15E+04 | 3.80E-06 | 1.77E-10 | 1.77E-08 |

| Radionuclide | 10,000-Year Disposal Limit (Ci/Vault 4)* | Vault 4 Projected Inventory (Ci)** | Fraction of 10,000-Year Disposal Limit | Dose (mrem/yr) |
|---|---|---|---|---|
| U-232 | 9.00E+03 | 9.52E-03 | 1.06E-06 | 1.06E-04 |
| U-233 | 1.35E+04 | 9.82E-01 | 7.27E-05 | 7.27E-03 |
| U-234 | 4.48E+03 | 6.59E+00 | 1.47E-03 | 1.47E-01 |
| U-235 | 1.03E+05 | 7.41E-02 | 7.19E-07 | 7.19E-05 |
| U-236 | 3.17E+08 | 1.42E-01 | 4.48E-10 | 4.48E-08 |
| U-238 | 6.60E+04 | 1.61E-01 | 2.44E-06 | 2.44E-04 |
| Np-237 | 6.73E+04 | 5.76E-01 | 8.56E-06 | 8.56E-04 |
| Pu-238 | 1.27E+07 | 3.69E+03 | 2.91E-04 | 2.91E-02 |
| Pu-239 | 1.37E+10 | 3.36E+01 | 2.45E-09 | 2.45E-07 |
| Pu-240 | 2.96E+12 | 8.39E+00 | 2.83E-12 | 2.83E-10 |
| Pu-241 | 1.02E+10 | 1.72E+02 | 1.69E-08 | 1.69E-06 |
| Pu-242 | 4.91E+10 | 9.32E-03 | 1.90E-13 | 1.90E-11 |
| Pu-244 | 3.65E+03 | 9.38E-06 | 2.57E-09 | 2.57E-07 |
| Am-241 | 3.38E+08 | 1.44E+01 | 4.25E-08 | 4.25E-06 |
| Am-242m | 9.83E+06 | 7.25E-03 | 7.38E-10 | 7.38E-08 |
| Am-243 | 2.96E+05 | 6.22E-03 | 2.10E-08 | 2.10E-06 |
| Cm-242 | 2.51E+09 | 6.21E-03 | 2.47E-12 | 2.47E-10 |
| Cm-243 | 7.00E+09 | 2.88E-03 | 4.11E-13 | 4.11E-11 |
| Cm-244 | 1.08E+15 | 3.16E+00 | 2.93E-15 | 2.93E-13 |
| Cm-245 | 8.42E+06 | 3.03E-04 | 3.60E-11 | 3.60E-09 |
| Cm-247 | 2.45E+04 | 5.55E-13 | 2.27E-17 | 2.27E-15 |
| Cm-248 | 4.64E+07 | 5.79E-13 | 1.25E-20 | 1.25E-18 |
| Bk-249 | 4.92E+07 | 4.23E-20 | 8.60E-28 | 8.60E-26 |
| Cf-249 | 1.27E+05 | 3.21E-12 | 2.53E-17 | 2.53E-15 |
| Cf-251 | 1.83E+06 | 2.47E-01 | 1.35E-07 | 1.35E-05 |
| Cf-252 | 6.31E+12 | 3.56E-15 | 5.64E-28 | 5.64E-26 |
| | | **Totals** | **2.17E-01** | **2.17E+01** |

* Vault 4 inventory limits from Table 3-2 of Cook et al., (2005) based upon intruder dose limit of 100 mrem/yr
** Projected inventory from d'Entremont and Drumm (2005)

## 6.0 10 CFR 61.43 Compliance (Protection of Individuals During Operations)

The performance objective in 10 CFR 61.43 cross-references "the standards for radiation protection in Part 20". The cross-referenced "standards for radiation protection" in 10 CFR Part 20 (USNRC, 2005) that are considered in detail in this Performance Objective Demonstration Document (PODD) Determination are the dose limits for the public and the workers during disposal operations set forth in 10 CFR 20.1101(d), 10 CFR 20.1201(a)(1)(i), 10 CFR 20.1201(a)(1)(ii), 10 CFR 20.1201(a)(2)(i), 10 CFR 20.1201(a)(2)(ii), 10 CFR 20.1201(e), 10 CFR 20.1208(a), 10 CFR 20.1301(a)(1), 10 CFR 20.1301(a)(2), and 10 CFR 20.1301(b). As will be discussed in the following sections, these dose limits correspond to the dose limits in 10 CFR Part 835 and relevant DOE Orders which establish DOE regulatory and contractual requirements for DOE facilities and activities. The following subsections show that disposal operations at SDF for the solidified salt waste streams meet these dose limits and that doses will be maintained as low as reasonably achievable.

### 6.1 Air Emissions Limit for Individual Member of the Public (10 CFR 20.1101(d))

The NRC regulation at 10 CFR 20.1101(d) provides in relevant part:

> [A] constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established … such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions.

DOE similarly limits doses from air emissions to the public to 10 mrem per year in DOE Order 5400.5. DOE is also subject to and complies with the Environmental Protection Agency requirement in 40 CFR 61.92, which has the same limit. The estimated dose per year from airborne emissions to the maximally exposed individual member of the public located at or beyond the SRS site boundary from all operations at SRS ranged from 0.04 mrem to 0.07 mrem from 1999 through 2003 (WSRC, 1999)(WSRC, 2000)(WSRC, 2001)(WSRC, 2002)(WSRC, 2003a). These values for all of the SRS operations (not just disposal operations at SDF) are well below the dose limit specified in 10 CFR 20.1101(d). DOE anticipates that the dose from air emissions from SDF following the receipt of the low-activity salt waste streams associated with DDA, ARP/MCU, and SWPF operations will be a small fraction of the dose from all operations at SRS.

## 6.2 Total Effective Dose Equivalent (TEDE) Limit for Adult Workers (10 CFR 20.1201(a)(1)(i))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

(a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.

(1) An annual limit, which is the more limiting of –

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv);

***

DOE's regulation at 10 CFR 835.202 (a)(1) has the same annual dose limit for the annual occupational dose to general employees[3]. For the occupational dose to adults during disposal operations at SDF, the TEDE per year will be controlled as low as reasonably achievable below 5 rem (WSRC, 2005b). In this regard, occupational doses to workers have been well within the annual limits specified in 10 CFR 20.1201 (2) for all work activities at SRS (200-Z-0001, 2003). The highest dose received by an SRS worker in 2003 was 1808 mrem TEDE. There was close to zero total exposure received by the SRS workforce for the SPF and SDF activities. Since 1998, the highest dose received by an SRS worker has been at or below 1808 mrem per year. The total dose received by workers at SPF and SDF since 1998 is 35 mrem, which is well below the limit specified in 10 CFR 20.1201(a). Furthermore, operations will continue to consist predominantly of mechanical mixing of low-activity salt solutions with cementitious material and then mechanical transfer to SDF; therefore, little hands-on work at SDF is anticipated. Thus, the total effective dose equivalent to workers from disposal of the solidified low-activity salt waste at SDF is expected to remain well below the NRC limit.

## 6.3 Any Individual Organ or Tissue Dose Limit for Adult Workers (10 CFR 20.1201(a)(1)(ii))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

[3] DOE's regulation requires that the occupational dose per year for general employees shall not exceed both a total effective dose equivalent of 5 rems and the sum of the deep dose equivalent for external exposures and the committed dose equivalent to any other organ or tissue other than the lens of the eye of 50 rems. NRC's regulation specifies that either of these two limits shall be met by NRC licensees, whichever is more limiting. This document will show that DOE will meet the more stringent of the dose limits in 10 CFR Part 835 and the relevant dose limits in 10 CFR Part 20. Because DOE imposes stricter, separate requirements, the provisions of 10 CFR 20.1201(a)(1) and (a)(2), which correlate to 10 CFR 835.202(a)(1) and 10 CFR 835.202(a)(2)), are discussed in separate subsections in this document.

(a)[C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.

(1) An annual limit, which is the more limiting of –

***

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

The dose limit specified in 10 CFR 20.1201(a)(1)(ii) is the same as that specified in 10 CFR 835.202 (a)(2). For the occupational dose to adults during disposal operations at SDF, the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye will be controlled to as low as reasonably achievable below a maximum of 50 rem per year (WSRC, 2005b). SRS Engineering Standard 01064, *Radiological Design Requirements* (WSRC, 2005a), provides that the design basis annual occupational exposure limits for any organ or tissue other than the eye cannot exceed 10 rem per year, which is well below the NRC limit of 50 rem per year. Furthermore, operations will predominantly consist of mechanical mixing of low-activity salt solutions with cementitious material and then mechanical transfer to SDF, so that little hands-on work, and little, if any, associated exposure, is anticipated for disposal operations in SDF.

## 6.4 Annual Dose Limit to the Lens of the Eye for Adult Workers (10 CFR 20.1201(a)(2)(i))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

(a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.

***

(2) The annual limits to the lens of the eye, to the skin of the whole body or to the skin of the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv)

The dose limit specified in 10 CFR 20.1201(a)(2)(i) is the same as that specified in DOE's regulation at 10 CFR 835.202 (a)(3). For the occupational dose to adults during

disposal operations at SDF, the annual dose limit to the lens of the eye will be controlled to as low as reasonably achievable below maximum of 15 mrem per year (WSRC, 2005b). SRS Engineering Standard 01064, *Radiological Design Requirements* (WSRC, 2005a), provides that the design basis annual occupational exposure limits for the lens of the eye cannot exceed 3 rem per year, which is well below the NRC limit of 15 rem per year. Furthermore, as noted previously, operations predominantly consist of mechanical mixing of low-activity salt solutions with cementitious material and then mechanical transfer to SDF; therefore, little hands-on work or exposure is anticipated during disposal operations at SDF.

## 6.5 Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers (10 CFR 20.1201(a)(2)(ii))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.
>
> ***
>
> (2) The annual limits to the lens of the eye, the skin of the whole body, or to the skin of the extremities, which are:
>
> ***
>
> (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

This NRC dose limit specified in 10 CFR 20.1201(a)(2)(ii) is the same as the DOE dose limit specified in 10 CFR 835.202 (a)(4). For the occupational dose to adults during disposal operations at SDF, which involve little hands-on activity, the annual dose limit to the skin of the whole body or to the skin of any extremity will be controlled to as low as reasonably achievable below a shallow-dose equivalent of 50 rem per year (WSRC, 2005b).

## 6.6 Limit on Soluble Uranium Intake (10 CFR 20.1201(e))

The NRC regulation at 10 CFR 20.1201(e), concerning occupational dose limits for adults, provides in relevant part:

> (e) In addition to the annual dose limits, … limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity [.]

In addition to the annual dose limits to adults during disposal operations at SDF, the soluble uranium intake by an individual is controlled to less than 10 milligrams (mg) per week. DOE Order 440.1A requirements for soluble uranium intake are the more restrictive of the concentrations in the American Conference of Governmental Industrial Hygienists Threshold Limit Values (0.2 mg per cubic meter which is the same as noted in 10 CFR 20 Appendix B footnote 3) or the Occupational Safety and Health Administration (OSHA) Permissible Exposure Limit (PEL) (0.05 mg per cubic meter) (USDOE, 1998). The OSHA PEL limit for soluble uranium, which equates to a soluble uranium intake of 2.4 mg per week, is the more restrictive of the two and therefore is the limit imposed for disposal operations at SDF. Accordingly, the soluble uranium intake, if any, during disposal operations at the SDF will be controlled to 2.4 mg per week, which is below the NRC limit in 10 CFR 20.1201(e). Moreover, the low-activity waste disposed of in the SDF will be in a solid form, and thus there will not be soluble uranium in the solidified waste susceptible to worker intake. In addition, as noted previously, disposal operations will generally be performed mechanically, without worker exposure during normal mechanical operations.

### 6.7 Dose Equivalent to an Embryo/Fetus (10 CFR 20.1208(a))

The NRC regulation at 10 CFR 20.1208(a), concerning the dose equivalent to an embryo/fetus, provides in relevant part:

> (a) [E]nsure that the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv).

DOE's regulation at 10 CFR 835.206 (a) has the same dose limit. For the occupational dose to an embryo/fetus during disposal operations at SDF, doses will be controlled so that the dose equivalent to the embryo/fetus during the entire pregnancy for a declared pregnant worker will not exceed 0.5 rem (WSRC, 2005b). Furthermore, after declaration of pregnancy, DOE provides the option of a mutually agreeable assignment of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure during the remainder of the gestation period is unlikely. In addition, personnel dosimetry is provided and used to carefully track exposure.

### 6.8 Dose Limits for Individual Members of the Public (10 CFR 20.1301(a)(1))

The NRC regulation at 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

> (a) [C]onduct operations so that -

(1) The total effective dose equivalent to individual members of the public …does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released…., from voluntary participation in medical research programs, and from the …disposal of radioactive material into sanitary sewerage [.]

DOE Order 5400.5 II.1.a similarly limits public doses to less than 100 mrem per year. However, DOE's application of the limit is more restrictive in that it requires DOE to make a reasonable effort to ensure that multiple sources (e.g., DOE sources and NRC regulated sources) do not combine to cause the limit to be exceeded. For individual members of the public during disposal operations at SDF, the TEDE limit to an individual member of the public will be controlled to less than 0.1 rem per year (WSRC,2005c). The air pathway is the predominate pathway for doses to the public from SRS operations, including disposal operations at SDF, and, as discussed in subsection 4.2 of this PODD, doses from the air pathway to members of the public have been, and are expected to continue to be, well below the 0.1 rem annual limit specified in 10 CFR 20.1301(a) (WSRC, 2005a).

### 6.9 Dose Limits for Individual Members of the Public (10 CFR 20.1301(a)(2))

The NRC regulation at 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

(a) [C]onduct operations so that –

* * *

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released …, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

DOE's regulation at 10 CFR 835.602 establishes the expectation that the TEDE in Controlled Areas will be less than 0.1 rem in a year. For individual members of the public during disposal operations at SDF, operations will be conducted such that the dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material, will be less than 0.00005 rem per hour above background. WSRC Manual 5Q, Chapter 2, Article 232, also restricts the TEDE in Controlled Areas to less than 0.1 rem in a year. To ensure that these dose limits are met, the following measures have been instituted within Controlled Areas. Per 10 CFR 835.603(g), Radioactive Materials Areas have been established for accumulations of radioactive material that could result in a radiation dose of 100 mrem in a year or greater. In addition, SRS has established Radiological Buffer Areas around posted Radiological

Areas. Standard SRS practice is to assume a 2,000 hour per year continuous occupancy at the outer boundary of these areas and; therefore, the dose rate at a Radiological Buffer Area boundary is 0.05 mrem/hr (0.1 rem/2,000 hr = 0.00005 rem/hr). Since the Controlled Area encompasses a Radiological Buffer Area, it is ensured that the dose in the Controlled Area (but outside of Radioactive Material Areas and Radiological Buffer Areas) will be less than 0.1 rem in a year (WSRC, 2005c). Therefore, SRS implementation of the provisions at 10 CFR 835.602 and 10 CFR 835.603 limit provides a more restrictive limit than the dose limit specified in 10 CFR 20.1301(a)(2). Furthermore, training is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem per year.

### 6.10 Dose Limits for Individual Members of the Public (10 CFR 20.1301(b))

The NRC regulation at 10 CFR 20.1301(b), concerning dose limits for individual members of the public, provides in relevant part:

> (b) If … members of the public [are permitted] to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

DOE's regulation at 10 CFR 835.208 has the same dose limit. The TEDE limit to an individual member of the public granted access to controlled areas[4] during disposal operations at SDF will be controlled to 0.1 rem per year (WSRC, 2005b). Furthermore, training is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem per year[5].

### 6.11 As Low As Reasonably Achievable (10 CFR 20.1003)

The NRC regulation at 10 CFR 20.1003 defines ALARA in relevant part:

> ALARA … means making every reasonable effort to maintain exposures to radiation as far below the dose limits … as is practical consistent with the purpose for which the … activity is undertaken…[.]

[4] 10 CFR 20.1003 defines restricted areas as an area, access to which is limited … for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. This is the same as the definition in 10 CFR 835.2 for a controlled area.

[5] 10 CFR 20.1301(d) allows licensees to request NRC authorization to allow an individual member of the public to operate up to an annual dose limit of 0.5 rem (5 mSv). 10 CFR 835 is more restrictive for the dose to an individual member of the public with a limit of 0.1 rem maximum annual dose as discussed in Subsection 5.8. Therefore, this limit is not discussed further.

DOE has a similar requirement, and DOE's regulation at 10 CFR 835.2 defines ALARA as "… the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable…". For radiological work activities during disposal operations at SDF, every reasonable effort will be made to maintain exposures to radiation as far below the dose limits as is practical consistent with the purpose for which the activity is undertaken. Furthermore, DOE's regulation at 10 CFR 835.101(c) requires the contents of each Radiation Protection Program (RPP) to include formal plans and measure for applying the as-low-as-reasonably-achievable process to occupational exposure. SRS ensures ongoing activities are evaluated by the tracking and publication of monthly worker dose reports (Freeman, 2005a). SRS also maintains doses ALARA by setting annual administrative control limits on worker dose that are significantly below the Federal limit of 5 rem. For 2005, the SRS annual administrative control limit is 0.8 rem (Freeman, 2005b).

### 6.12 Reasonable Assurance

Measures that provide reasonable assurance that disposal operations at SDF will comply with the applicable dose limits and with the ALARA provisions include: (1) the documented RPP; (2) the Documented Safety Analysis (DSA); (3) design; (4) regulatory and contractual enforcement mechanisms; and (5) access controls, training, and dosimetry. These measures are discussed in the following paragraphs. In addition, the following discusses the exposure history at SRS and at SDF.

### 6.13 SRS Radiation Protection Program

The Department of Energy regulates occupational radiation exposure at its facilities through 10 CFR Part 835, *Occupational Radiation Protection*. Part 835 establishes exposure limits and other requirements to ensure that DOE facilities are operated in a manner such that occupational exposure to workers is maintained within acceptable limits and as far below these limits as is reasonably achievable. The requirements in Part 835 are nuclear safety requirements which, if violated, provide a basis for the assessment of civil penalties under the section 234A of the Atomic Energy Act.

Pursuant to Part 835, activities at SRS, including disposal operations at SDF, must be conducted in compliance with the documented RPP for SRS as approved by DOE. The key elements of the RPP include monitoring of individuals and work areas, control of access to areas containing radiation and radioactive materials, use of warning signs and labels, methods to control the spread of radioactive contamination, radiation safety training, objectives for the design of facilities, criteria for levels of radiation and radioactive material in the workplace, and continually updated records to document

compliance with the provisions of Part 835. The RPP also includes formal plans and measures for applying the ALARA process.

The requirements of Part 835 as contained in the approved RPP are incorporated in the WSRC's Standards and Requirements Implementing Document system. The Standards and Requirements Implementing Document system links the requirements of Part 835 to the site-level and lower-level implementing policies and procedures that control radiological work activities conducted across the site. These requirements are primarily contained in the WSRC 5Q Manual, *Radiological Control*, and its lower tier manuals, e.g., WSRC 5Q1.1, *Radiation and Contamination Control Procedures Manual*, and WSRC 5Q1.2, *Radiation Monitoring Procedures Manual*. These procedures control the planning of radiological work, the use of radiation monitoring devices by employees, the bioassay program, the air monitoring program, the contamination control program, the ALARA program, the training of general employees, radiological workers, Radiological Control Inspectors, and health physics professionals and technicians, and the other aspects of an occupational radiation protection program as required by Part 835.

## 6.14 Documented Safety Analysis

A DSA (WSRC, 2004a) has been approved by DOE for operation of SPF and SDF in accordance with 10 CFR Part 830. As the first step in the development of the DSA, a formal Consolidated Hazards Analysis (CHA) (WSRC, 2004b) was performed at the Saltstone Facility to evaluate the potential risk of operations to the workers and the public. The CHA was performed by a group of approximately 20 subject matter experts, with expertise in the fields of operations, engineering, industrial hygiene, radiological protection, environmental compliance, and maintenance.

The CHA consisted of three basis phases: hazard identification; hazard classification; and hazard evaluation. During the hazard identification phase, all possible radiological and chemical hazardous materials associated with the normal and abnormal operations of the facility were identified, along with all potential energy sources available to disperse the hazardous materials to the environment.

During the hazard classification phase, the maximum quantities of hazardous materials possible in the Saltstone Facility are evaluated against the criterion listed in DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, to determine the overall hazard classification of the facility. It was determined by the CHA team that the hazard classification of the Saltstone Facility was Hazard Category 3, which is the lowest hazard classification and denotes a potential for only localized consequences to workers at the facility and no potential for significant consequences to other workers at the site or to members of the public.

During the third and final phase of the CHA, all possible normal and abnormal operational events that could result in exposing facility workers or the public to hazardous material were evaluated to determine the magnitude of the risk. During this hazard evaluation phase, the consequence and frequency of each operational event was qualitatively determined, and the resulting level of risk was identified. The purpose of identifying the level of risk was to determine which operational events posed some level of risk (and thus required additional evaluation) and those events which presented negligible risk to the facility workers and public. As a result of the hazard evaluation for the Saltstone Facility, all normal operational events were determined to present negligible risk to the workers and public (i.e., exposure < 5 rem to facility workers), and were thus removed from further evaluation. For purposes of this CHA, the waste inventory and curie concentrations were assumed to be greater than currently planned for the DDA, ARP/MCU, and SWPF streams.

The DSA analyzed the hazards that were identified in the CHA that could impact facility workers during normal operations and accident conditions, and specifically included radiation exposure hazards. The DSA identified the basis for derivation of the Saltstone Facility Technical Safety Requirements (TSR) (WSRC, 2003b) and also discussed summary descriptions of the key features of safety management programs at SRS as they pertain to SDF.

The Saltstone Facility TSR document identified the administrative controls that are necessary to achieve safe operations at SDF. In part, these TSR administrative controls require: (1) that a facility manager be assigned who is accountable for safe operation and in command of activities necessary to maintain safe operation; (2) that personnel who carry out radiological controls functions for SDF have sufficient organizational freedom to ensure independence from operating pressures; (3) that SDF personnel receive initial and continuing training including radiological control training; and (4) that an RPP shall be prepared consistent with 10 CFR Part 835. The DSA determined that the administrative controls identified in the TSR are sufficient to ensure worker protection in accordance with 10 CFR Part 835.

In addition, the design requirements for SPF and SDF implemented 10 CFR Part 835 and, in particular, implemented ALARA principles. The design is currently being upgraded to reflect the radionuclide concentrations in the low-activity waste streams to be received at SPF and SDF from planned Interim Salt Processing facilities and SWPF. While the upgraded design is not yet complete, based on the current SPF and SDF design, it is estimated that occupational exposures for SPF and SDF workers will be at least an order of magnitude lower than the 10 CFR Part 835 dose limit of 5 rem per year during both Interim Salt Processing and SWPF operation.

### 6.15 Radiological Design for Protection of Occupational Workers and the Public

New SRS radiological facilities and facility modifications including the ongoing Saltstone Facility modifications are designed to meet the requirements of 10 CFR 835 Subpart K, *Design and Control* (USDOE, 2004). SRS Engineering Standard 01064, *Radiological Design Requirements* (WSRC, 2005a), provides the requirements necessary to ensure compliance with 10 CFR Part 835. The standard refers to 10 CFR Part 835, DOE orders, DOE standards, DOE handbooks, national consensus standards, SRS manuals, SRS engineering standards, SRS engineering guides, and site operating experience in order to meet the 10 CFR Part 835 specific requirements and additional requirements to ensure the design provides for protection of the worker and the environment.

The standard covers the full spectrum of radiological design requirements and not just radiation exposure limits. The following are the specific areas addressed in the standard: radiation exposure limits; facility and equipment layout; area radiation levels; radiation shielding; internal radiation exposure; radiological monitoring; confinement; and ventilation.

The design requirements for several of the important sections of the standard are highlighted in order to understand the design limits and philosophy for SRS designs. The first area of interest is the radiation exposure limits. The following is an excerpt of the standard which presents the exposure limits and philosophy for both external and internal radiation exposure.

> "During the design of new facilities or modification of existing facilities, the design objective for controlling personnel exposure from external sources of radiation in areas of continuous occupancy (2000 hours per year) shall be to maintain exposure levels below an average of 0.5 mrem per hour and as far below this average as is reasonably achievable. The design objectives for exposure rates for potential exposure to a radiological worker where occupancy differs from the above shall be As Low As Reasonably Achievable (ALARA) and shall not exceed the external limits in Table 5-1. Regarding the control of airborne radioactive material, the design objective shall be, under normal conditions, to avoid releases to the workplace atmosphere and in any situation, to control the inhalation of such material by workers to levels that are ALARA; confinement and ventilation shall normally be used [6.3, 6.12]. Table 5-1 summarizes the design basis external radiation exposure limits.

**[Table 5-1]**

Design Basis Annual Occupational Radiation Exposure Limits

| Type of Exposure | Limit (rem) |
|---|---|
| Whole Body TEDE | 1.0 |
| Internal CEDE | 0.5 |
| Lens of Eye | 3 |
| Extremity | 10 |
| Any Organ (other than eye) or Tissue | 10 |

To meet the Site's no deliberate intake policy, engineered controls will be evaluated and implemented to ensure that, under normal operating conditions, no worker will receive a deliberate intake of radionuclides (i.e., CEDE=0 rem). As a result, the TEDE limit will be independent of the CEDE limit. The 0.5 rem CEDE limit in Table 5.1 is to be applied to potential intakes from anticipated potential releases or anticipated off-normal maintenance. Under these anticipated potential conditions, engineered controls will be evaluated and implemented to minimize the potential for workers to receive intakes that will exceed the 0.5 rem CEDE. This evaluation will not take credit for the use of respiratory protection.

The dose to any member of the public or a minor exposed to radiation at a DOE facility shall not exceed 0.1 rem TEDE in a year."

The facility design also incorporates radiation zoning criteria in order to ensure the exposure limits presented above are met by providing adequate radiation shielding. Areas in which non-radiological workers are present are assumed to have continuous occupancy (2,000 hours per year) and are designed to a dose rate less than 0.05 mrem per hour to ensure that the annual dose is less than 100 mrem. Other zoning criteria are established to ensure radiological worker doses are ALARA and less than 1,000 mrem per year to meet the 10 CFR 835.1002 design requirements.

The design is also required to provide necessary radiological monitoring or sampling for airborne and surface contamination to ensure the engineered controls are performing their function and, in the event of a failure or upset condition, workers are warned and exposures avoided.

Radiological protection personnel ensure the requirements of the standard are addressed and presented in design summary documentation (White, 2003). The incorporation of all the radiological design criteria in the engineering standard ensures the requirements of 10 CFR Part 835 are met and the design provides for the radiological safety of the workers and environment.

## 6.16 Regulatory and Contractual Enforcement

Any violation of the requirements in 10 CFR Part 835 is subject to civil penalties pursuant to section 234A of the Atomic Energy Act of 1954, as amended, 42 USC 2011 et seq., as implemented by DOE regulations in 10 CFR Part 820. In addition, the requirements in 10 CFR Part 835 and all applicable DOE Orders are incorporated into all contracts with DOE contractors, including WSRC, the DOE contractor for disposal operations at SDF as well as other operations at SRS. DOE enforces these contractual requirements through contract enforcement measures, including the reduction of contract fees.

## 6.17 Access Controls, Training, Dosimetry, and Monitoring

Training or an escort is required for individual members of the public for entry into controlled areas. In addition, use of dosimetry is required if a member of the public is expected to enter a controlled area and exceed 0.05 rem per year to ensure no member of the public exceeds radiation exposure limits (WSRC, 2005d).

In addition, worker radiation exposure monitoring is performed for all workers expected to receive 100 mrem per year from internal and external sources of radiation to provide assurance that no worker exceeds radiation exposure limits and that all radiation dose are maintained as far below the limits as is reasonably achievable.

## 6.18 Occupational Radiation Exposure History for Savannah River Site

The effectiveness of the radiation protection programs, including the effectiveness of oversight programs to ensure they are implemented properly is demonstrated by the occupational radiation exposure results as documented for 2003 (Freeman, 2004). The highest dose received by an SRS worker in 2003 was 1808 mrem TEDE compared to the DOE Administrative Control Limit of 2000 mrem per year and the 10 CFR Part 835 limit of 5000 mrem per year. There was close to zero total exposure received by the SRS workforce for the SPF and SDF activities. Since 1998, the highest dose received by an SRS worker has been at or below 1808 mrem per year. The total dose received by workers at SPF and SDF since 1998 is 35 mrem.

In addition, for all work activities, the average TEDE exposure for workers receiving a TEDE dose at SRS has been 75 mrem per year or less since 2001 (US DOE, 2003). It is expected that exposures for workers at SDF will be at or below this average based on design requirements and past experience with SDF operation.

## 6.19 Conclusion

The information presented in this section provides assurance that the performance objective has been met for salt waste disposal. Therefore, the Saltstone Disposal Facility complies with the radiological protection of workers during operations performance objectives in 10 CFR 61.43.

# 7.0 10 CFR 61.44 Compliance (Long-Term Stability of the Disposal Site)

10 CFR 61.44 states:

> The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.

The SDF is currently an operational low-level waste disposal facility to which additional disposal vaults will be constructed on a just-in-time basis. The facility is not scheduled for closure until completion of salt waste disposition currently estimated for 2019. A final closure plan will be developed in the future, as appropriate, to support closure of SDF. As demonstrated below, SDF will meet the performance objective at 10 CFR 61.44 for long-term stability of the disposal site.

## 7.1 Siting

SDF is described in Section 3.0. Additional information pertinent to this performance objective follows.

Two major earthquakes have occurred within 100 miles of SRS. The largest known earthquake to affect SRS was the Charleston earthquake of 1886, with an epicenter approximately 90 miles from SRS and a magnitude of 6.6 on the Richter Scale. It is estimated that an earthquake of this magnitude would result in a peak ground acceleration of 0.10g at SRS (URS/Blume, 1982). A seismic evaluation of Z-Area shows that the soils beneath Z-Area are not susceptible to significant liquefaction for earthquakes having a peak ground acceleration less than or equal to 0.17g (McHood, 2002). The second earthquake occurred approximately 90 – 100 miles from SRS, with an estimated magnitude of 4.5.

Most of the soils in the vicinity of SDF are sandy over a loamy or clayey subsoil. The dominant vegetation near SDF is forest with types ranging from scrub oak to cypress with pine being the primary forest in the area (DOE/EIS-0082-S2, 2001). The siting of SDF, including the additional vaults where the solidified low-activity salt streams will be disposed, is such that it provides long-term stability consistent with this performance objective.

## 7.2 Design

SDF currently contains two large concrete vaults divided into cells. Each of the cells will be filled with solidified waste. The grout used to solidify the waste provides primary containment of the waste and the walls, floor, and roof of the vaults provide secondary containment. Approximately 15 feet of overburden have been removed to prepare and level the site for vault construction. All vaults will be built at or slightly below the grade level that exists after the overburden and leveling operations are complete. The bottom of the saltstone grout monoliths will be at least five feet above the historic high water table beneath SDF thus, avoiding disposal of waste in a zone of water table fluctuation.

Run-on and runoff controls are installed to minimize site erosion during the operational period. Approximately 160 acres are bounded by the Saltstone Facility perimeter fence. The impervious portion of this area is approximately 5% of the total Saltstone Facility property. Stormwater runoff from the impervious area of the Saltstone Facility is channeled through a pipe to the Z-01 sedimentation basin. McQueen Branch receives any discharges from the Z-01 basin. The basin is inspected at least once every three months (Because this is a relatively small drainage area, no special measures for runoff control are required (Sutherland, 2004).

The current active vault (Vault 4) description is given in section 2. Additional information for Vault 4 is presented below. The six cells within Vault 4 that will be used during Interim Salt Processing have a leachate collection system installed within the vault walls. This prevents hydraulic pressure build-up against the vault walls. A sheet drain system is installed on the cell walls with a 12" pipe at the bottom of the walls to collect the leachate. Each cell has a drain line that can be accessed from the exterior of the cells. Prior to Interim Salt Processing, modifications are being made to these cells to install a pump and piping system to transfer the collected leachate from the cells to the grout transfer line upstream of the cells. The leachate will be pumped at a low rate into the grout line for re-introduction to the cells during grout-fill operations.

The other existing vault (Vault 1) has the dimensions of approximately 100 feet wide, by 600 feet in length, by 25 feet in height. The vault is divided into six cells, with each cell measuring approximately 100 feet by 100 feet.

DOE is currently evaluating design alternatives for future SDF vaults. These new vaults will be designed to meet all of the same standards as the existing vaults. The new design will provide the same level or a greater level of protection against infiltration of water, migration of radioactive contaminants, structural integrity, and radiation shielding. New designs are being considered primarily to maximize the processing capacity of the facility, to simplify the operation of the vaults, and to minimize the cost of construction. Therefore, the existing vaults are, and the future vaults will be, designed to achieve the required long-term stability of the disposal site.

The Saltstone Facility was originally designed for processing materials with lower radionuclide concentrations than those currently planned during Interim Salt Processing. In preparation for Interim Salt Processing, several modifications in the design and operation of the facility are being undertaken. For example, several inches of grout which does not contain the low-activity waste stream will be poured on top of the low-level grout in the inactive vault cells to reduce the amount of exposure to facility workers (which will be primarily in the SPF rather than SDF portion of the facility) due to sky shine.

## 7.3 Use/Operation

SDF is permitted as an Industrial Solid Waste Landfill (ISWLF) site, as defined by SCDHEC Regulations R61-66 and R.61-107.16. Active disposal operations are planned to occur at SDF until around 2019. Following the filling of each vault cell, monitoring of the vaults occurs. Except for erosion control purposes, backfilling around the vaults will likely not be done prior to the completion of disposal activities. This will allow the vaults to be visually monitored for several years prior to closure. Any liquid that may accumulate as a result of rain water infiltration will be drained and returned to the process to avoid creating bulges in the vault walls. Routine surveillances of the filled vaults for structural integrity and soundness are conducted (200-Z-00001, 2003).

## 7.4 Closure

The stability of the SDF closure design is an important element for meeting SDF performance objectives. The SDF facility design is focused on minimizing the contact of water with the stabilized waste form. The saltstone grout material in the stabilized waste form will help retard the migration of rainwater to the waste. The existing storm water control systems will be maintained (including monitoring, surveillance, and minor custodial care activities). To further ensure long-term stability of SDF, the land in Z-Area which includes SDF, will remain under the ownership of the Federal Government. The three counties making up SRS have zoning restrictions that prevent the purchase of property or the approval of building permits at SRS. SRS is zoned "Department of Energy ownership". Residential use of this land will be prohibited via continued land use leasing restrictions (USDOE, 2000). No unrestricted use of the land or groundwater will be permitted for SDF.

Closure operations will begin near the end of the active SDF disposal period, in approximately 2019. Although final closure plans for SDF will be developed in the future, the actions that are contemplated today are given in Phifer and Nelson, 2003. Figure 3-5 gives a pictorial view of the closure plan. SDF will be closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site so that only surveillance, monitoring, or minor custodial care are required.

### 7.6 Results of the Long-Term Stability of the Disposal Site

The information presented in this section provides assurance that the performance object has been met for the salt waste disposal. Thus, the Saltstone Disposal Facility complies with the NRC long-term stability performance objective in 10 CFR 61.44.

# 8.0 Sensitivity Analysis and Uncertainty Analysis

The objective of the performance assessment calculations is to quantitatively estimate the system performance for comparison to the performance objectives of 10 CFR 61, Subpart C. The sensitivity analyses identify the assumptions and parameters that affect the quantitative estimate of performance by evaluating the effects of changing the values of input variables or changing model structures. The uncertainty analysis provides a tool for understanding, in quantitative terms, the effect of parameter and model uncertainties. These uncertainties are described by considering a reasonable range of conditions, processes, or events to test the robustness of the SDF in comparison to the performance objectives.

The sensitivity and uncertainty analysis has been expanded for the current radiological composition of the waste to demonstrate that compliance with the performance objectives of 10 CFR 61, Subpart C can be reasonably assured.

## 8.1 Sensitivity of Groundwater Model Parameters

The result of the all-pathways analysis in this PODD is dose (mrem/yr). Therefore, factors that affect the estimation of dose are the focus of the sensitivity and uncertainty analysis discussed in this section. These factors include those that influence the transportation of radionuclides through saturated and unsaturated media (soil, concrete, saltstone, etc).

A series of Saltstone Vault 4 sensitivity calculations were performed using PORFLOW (ACRI, 2002) to quantify the impact of key model parameter settings on groundwater contaminant concentrations and dose at the 100 meter compliance hypothetical well using a time frame at the conclusion of institution control (IC) through 10,000 years. Four radionuclides, H-3, C-14, Se-79, and I-129; were chosen as the limiting cases for this sensitivity analysis because the 2005 SA shows that these radionuclides are the major contributor to the dose when all pathways are considered. Parameters for the scenarios are identified in Tables 8-1 and 8-2 and are described in detail in the following sections.

### 8.1.1 Key Model Parameters

#### 8.1.1.1 Infiltration rates through the upper geosynthetic clay liner

The changes in infiltration rates through the upper GCL are reflected by different land use scenarios. Three land use scenarios were modeled. The nominal case assumes the land use scenario of a 100 year institutional control (IC) period, which is bamboo cover, followed by development of a pine forest cover. The second land use scenario is a continuous bamboo cover. The third land use scenario is a 100 year institutional control followed by farming and eventually development of a pine forest cover. These different

land use scenarios impact the effectiveness and longevity of the Vault 4 closure cap, identified as the infiltration rate. The different infiltration rates also impact the hydraulic properties of the lower drainage layer and the vault base drainage layer due to transport and accumulation of silt in the drainage layers. Base case (nominal), lower and upper bounding infiltration through the upper GCL is provided by Phifer (2005). Higher infiltration rate through the upper GCL results in higher transport rates of silt through the drainage layers and more rapid accumulation of silt.

#### 8.1.1.2 Saltstone Waste Form and the Vault Concrete Parameters

The fundamental concept of the SDF is the controlled radionuclide release. Due to the low hydraulic conductivity and low molecular diffusion in cementitious materials, contaminant leaching from the SDF is very slow. The hydraulic conductivities represent the ease through which the water will pass through the material. There are two parts to the hydraulic conductivity, the initial saturated hydraulic conductivity and the hydraulic conductivity rate. The hydraulic conductivity rates are expressed in terms of a degradation rate constant ($\alpha$). The rates at which the hydraulic conductivities of the Saltstone waste form and the Saltstone vault concrete increase over time were varied around the values used in the 2005 SA (the 2005 SA value is considered the nominal value). The higher the hydraulic conductivity, the higher the degradation rate constant, and the faster the degradation of the material occurs.

The initial saturated hydraulic conductivity of the material used in the 2005 SA is identified as Ksat. For the Saltstone waste form the Ksat is $10^{-12}$ cm/sec. For the vault concrete Ksat is $10^{-11}$ cm/sec. For the sensitivity analysis, the Ksat's were varied by an order of magnitude about these values.

For the purposes of this sensitivity analysis, the relative permeability was set to unity thus forcing the vault and saltstone grout to be saturated through the analysis period.

For the purposes of this sensitivity analysis, the molecular diffusion coefficients were varied by an order of magnitude about the values used in the 2005 SA.

For the purposes of this sensitivity analysis, the distribution coefficients ($K_d$) were set to zero for these species in the vadose and aquifer transport simulations. (H-3 is zero in the base case.)

### 8.1.2 Scenario Description and Input Parameters

Table 8-1 summarizes the scenario runs and the corresponding sensitivity setting of each key modeling parameter. Scenario run 1 represents the nominal or base case for each contaminant species. The nominal designation shown in Tables 8-1 and 8-2 refer to the value of the model parameter setting used in the 2005 SA. The sensitivity runs include scenario runs 2 through 19. The paragraphs following the tables discuss what the different scenario runs represent and a basis for selection of the parameter setting.

**Table 8-1. Sensitivity Scenarios and Settings for Infiltration, Vadose Zone Concrete and Saltstone Hydraulic Conductivity**.

| Run | Infiltration | Vadose Zone Concrete Hydraulic Conductivity | Vadose Zone Saltstone Hydraulic Conductivity | Distribution Coefficient | Vadose Zone Concrete Diffusion Coefficient ($D_M$) | Vadose Zone Saltstone Diffusion Coefficient ($D_M$) |
|---|---|---|---|---|---|---|
| 1 | IC to Pine Forest | Nominal | Nominal | Nominal | Nominal | Nominal |
| 2 | Continuous Bamboo Cover | Nominal | Nominal | Nominal | Nominal | Nominal |
| 3 | IC to Farm to Pine Forest | Nominal | Nominal | Nominal | Nominal | Nominal |
| 4 | IC to Pine Forest | $\alpha = 1.0$ | Nominal | Nominal | Nominal | Nominal |
| 5 | IC to Pine Forest | $\alpha = 2.0$ | Nominal | Nominal | Nominal | Nominal |
| 6 | IC to Pine Forest | 0.1×Ksat | Nominal | Nominal | Nominal | Nominal |
| 7 | IC to Pine Forest | 10×Ksat | Nominal | Nominal | Nominal | Nominal |
| 8 | IC to Pine Forest | Nominal | $\alpha = 0.5$ | Nominal | Nominal | Nominal |
| 9 | IC to Pine Forest | Nominal | $\alpha = 1.5$ | Nominal | Nominal | Nominal |
| 10 | IC to Pine Forest | Nominal | 0.1×Ksat | Nominal | Nominal | Nominal |
| 11 | IC to Pine Forest | Nominal | 10×Ksat | Nominal | Nominal | Nominal |
| 12 | IC to Farm to Pine Forest | $\alpha = 2.0$ | $\alpha = 1.5$ | Nominal | Nominal | Nominal |
| 13 | IC to Pine Forest | $k_r = 1$ | $k_r = 1$ | Nominal | Nominal | Nominal |
| 14 | IC to Pine Forest | Nominal | Nominal | Nominal | $0.1 \times D_M$ | Nominal |
| 15 | IC to Pine Forest | Nominal | Nominal | Nominal | $10 \times D_M$ | Nominal |
| 16 | IC to Pine Forest | Nominal | Nominal | Nominal | Nominal | $0.1 \times D_M$ |
| 17 | IC to Pine Forest | Nominal | Nominal | Nominal | Nominal | $10 \times D_M$ |
| 18 | IC to Pine Forest | Nominal | Nominal | Nominal | $10 \times D_M$ | $10 \times D_M$ |
| 19 | IC to Pine Forest | Nominal | Nominal | 0 | Nominal | Nominal |

IC – Institutional Control
$\alpha$ = degradation rate constant
$k_r$ = relative permeability

The sensitivity of the all-pathways dose to land use and closure cover degradation is captured in Sensitivity Scenario Runs 2 and 3. The sensitivity to different land use scenarios above the Vault 4 is captured by Sensitivity Scenario Runs 2 and 3. The nominal case (Sensitivity Scenario Run 1) assumes a 100 year institutional control (bamboo cover) period followed by development of a pine forest cover. Sensitivity Scenario Run 2 is a land use scenario with a continuous bamboo cover. Sensitivity Scenario Run 3 is a land use scenario with a 100 year institutional control followed by farming and eventually development of a pine forest cover. Figure C-1 in Appendix C shows the infiltration rate through the upper GCL for the three different land use scenarios. The different infiltration rates also impact the hydraulic properties of the lower drainage layer and the vault base drainage layer due to transport and accumulation of silt in the drainage layers. Higher infiltration rate through the upper GCL results in higher transport rates of silt through the drainage layers and more rapid accumulation of silt. The variation over time of the saturated horizontal conductivity of the lower drainage layer and the vault base drainage layer is shown in Appendix C, Figures C-2 and C-4, respectively. Similarly, the variation over time of the saturated vertical conductivity of the lower drainage layer and the vault base drainage layer is shown in Appendix C, Figures C-3 and C-5, respectively. In both cases, there is a substantial reduction in the performance of the horizontal drainage layers over time due to the accumulation of silt.

The sensitivity of the all-pathways dose to the degradation of the concrete vault is captured in Sensitivity Scenario Runs 4, 5, 6, and 7. Sensitivity Scenario Runs 4 and 5 address the rate at which the concrete vault saturated hydraulic conductivity increases with time due to degradation of the concrete as the result of chemical attack or cracking. The nominal variation over time for the saturated hydraulic conductivity of the concrete vault is assumed to increase by three orders of magnitude after the 100 year IC period through 10,000 years as shown in Appendix C, Figure C-6. The functional form for the increase in conductivity over time is documented in the 2005 SA and is based on engineering judgment. Sensitivity Scenario Runs 4 and 5 increase the concrete vault conductivity by two and four orders of magnitude, respectively. In Sensitivity Scenario Runs 6 and 7, the concrete vault saturated hydraulic conductivity sensitivity is addressed. The increase in concrete vault saturated hydraulic conductivity over time, due to degradation, is taken from the 2005 SA and is used as the nominal rate. For the Sensitivity Scenario runs, the concrete vault saturated hydraulic conductivity is varied by an order of magnitude about the nominal value over the entire simulation period Appendix C, Figure C-7 shows the nominal condition and the sensitivity values.

The sensitivity of the all-pathways dose to the degradation of the Saltstone waste form is captured in Sensitivity Scenario Runs 8, 9, 10, and 11. Sensitivity Scenario Runs 8 and 9 address the rate at which the Saltstone saturated hydraulic conductivity increases with time due to degradation of the Saltstone waste form as the result of chemical attack or cracking. The nominal variation over time for the Saltstone waste form is assumed to increase by two orders of magnitude after the 100 year IC period through 10,000 years, as is shown in Appendix C, Figure C-8. Sensitivity Scenario Runs 7 and 8 increase the

conductivity by one and three orders of magnitude, respectively. In Sensitivity Scenario Runs 10 and 11, the Saltstone waste form saturated hydraulic conductivity is varied by an order of magnitude about the nominal value over the entire simulation period. The nominal rate of increase in conductivity, due to degradation over time, is used. Appendix C, Figure C-9 shows the nominal condition and the sensitivity values.

The sensitivity of the all-pathways dose to combination effect of high filtrations with degraded horizontal drain performance is captured in Sensitivity Scenario Run 12. Sensitivity Scenario Run 12 is a combined sensitivity based on s Sensitivity Scenario Runs 3, 5 and 9. A high infiltration rate with degraded horizontal drain performance (Sensitivity Scenario Run 3) is combined with highest rate increase in saturated hydraulic conductivity of the concrete vault and Saltstone over time.

The sensitivity of the all-pathways dose to uncertainties in the water retention curves for the concrete vault and Saltstone is captured in Sensitivity Scenario Run 13. The relative permeability ($k_r$) of the concrete vault and the Saltstone waste form was set to unity in Sensitivity Scenario Run 13. This was done to address uncertainties in the water retention curves for the concrete vault and Saltstone.

The sensitivity of the all-pathways dose to uncertainties in the diffusion coefficient for each radionuclide as they pass through the Saltstone waste form and the concrete vault are captured in Sensitivity Scenario Runs 14, 15, 16, 17, 18, and 19. The nominal values of the molecular diffusion coefficients for each species in the concrete vault are shown in Table A-9 of the 2005 SA (Cook et al., 2005). The molecular diffusion coefficients for each species in the concrete vault were varied an order of magnitude about their nominal values for Sensitivity Scenario Runs 14 and 15. The molecular diffusion coefficients for each species in the Saltstone waste form were varied an order of magnitude about their nominal values shown in Table A-9 of the 2005 SA (Coo et al, 2005). Sensitivity Scenario Runs 16 and 17 address this sensitivity. Sensitivity Scenario Run 18 is a combined sensitivity run of Sensitivity Scenario Runs 15 and 17. The molecular diffusion coefficients for each species are an order of magnitude higher than nominal for both the concrete vault and the Saltstone waste form in this scenario.

A distribution coefficient of zero was used through out the vadose and aquifer zone transport simulations for C-14, I-129 and Se-79 for Scenario 19. This case is not considered credible, but it does show the importance of the distribution coefficient in the model calculations.

### 8.1.3 Sensitivity Results

The predicted peak fractional fluxes to the water table and peak concentrations for C-14, H-3, I-129 and Se-79 are shown in Appendix C, Tables C-1 to C-4, respectively. All the radionuclides except H-3 appear to show the logical trend of lower/higher peak concentration with lower/higher sensitivity setting for a given parameter.

In Appendix C, Table C-2, the nominal case (scenario 1) for H-3 has a higher peak concentration than scenario 3 as a result of a higher infiltration rate over the first 800 years.

The H-3 peak concentration appears to be insensitive to changes in the concrete vault and Saltstone saturated hydraulic conductivity over the ranges assumed in the sensitivity analysis.

#### 8.1.3.1 Sensitivity Results Expressed as Dose from All Pathways

The peak fractional concentrations and the revised inventory of radionuclides in Vault 4 were used to calculate peak radionuclide concentrations over 10,000 years. The peak concentrations were input to the LADTAP program (Simpkins, 2004a) to calculate the all pathways dose for each of the scenarios. The resulting doses are shown in Table 8-2. The doses range from 0.02 mrem/year for scenario 2 (decreased infiltration due to continuous bamboo cover) to 38 mrem/year for scenario 19 (an incredible case in which all radionuclide distribution coefficients set to zero).

**Table 8-2. All-Pathways Doses from the Sensitivity Scenarios.**

| Scenario Run | Dose (mrem/year) |
|---|---|
| 1 | 5.12E-02 |
| 2 | 2.12E-02 |
| 3 | 2.81E-01 |
| 4 | 3.31E-02 |
| 5 | 5.42E-02 |
| 6 | 3.36E-02 |
| 7 | 5.39E-02 |
| 8 | 3.97E-02 |
| 9 | 2.47E-01 |
| 10 | 4.00E-02 |
| 11 | 2.57E-01 |
| 12 | 4.18E+00 |
| 13 | 1.87E-01 |
| 14 | 3.66E-02 |
| 15 | 1.83E-01 |
| 16 | 4.16E-02 |
| 17 | 6.74E-02 |
| 18 | 7.15E-01 |
| 19 | 3.78E+01 |

#### 8.1.3.2 Sensitivity of Technetium-99 to Oxidation/Reduction State

Three additional sensitivity runs were made to explore the sensitivity of Tc-99 to its oxidation/reduction state in saltstone grout and the vault concrete. Technetium scenario run A uses the nominal settings for all parameters in the 2005 SA, similar to scenario run 1 as outlined in Table 8-1. Technetium scenario run B reduced the $K_d$ for Tc-99 in Saltstone waste form and the vault concrete from 1000 to 1 mL/g, the value recommended for oxidizing concrete (Bradbury and Sarott, 1995). Technetium scenario run C reduced the $K_d$ for Tc-99 to zero, an incredibly pessimistic value. The results are shown in Appendix C, Table C-5. Using the projected Vault 4 inventory (Table 3-2) and the LADTAP program, these results can be expressed as doses. The Tc-99 dose from run A is 1.70E-13 mrem/year, that from run B is 3.36E+00 mrem/year and that from run C is 9.54E+01 mrem/year. Work on the loss of reducing capacity in Saltstone has shown that after 10,000 years 3% of the Saltstone will have become oxidized (Kaplan and Hang, 2003). Assuming that the 3% of the Saltstone is oxidized at time zero, the doses from Scenarios B and C can be approximated by taking 3% of the doses for Scenarios B and C, 0.10 and 2.9 mrem/year, respectively.

#### 8.1.3.3 Sensitivity to Vault Radionuclide Inventory

The sensitivity of the groundwater all-pathways dose to the inventory of radionuclides in Vault 4 was considered. The remaining available volume in Vault 4 will accommodate all of salt waste batches 0 through 7 and about half of batch 8. Two hypothetical vault 4 inventories were developed by assuming that two additional vaults the same size as Vault 4 would be built and would receive the salt waste after Vault 4 was filled. The first of these two hypothetical vaults, designated Vault X, would receive the remaining half of salt waste batch 8 (d'Entremont and Drumm, 2005), all of batch 9 and the remaining 11.9 million gallons would be SWPF waste. The second vault, designated Vault Y, would receive only SWPF waste.

The peak fractional concentrations for the nominal case (i.e., scenario run 1 for H-3, C-14, Se-79, and I-129 and technetium scenario run 1) were converted to doses using the three vault inventories as outlined above and the LADTAP program. The results are a dose of 5.12E-02 mrem/year for Vault 4, 2.85E-01 mrem/year for Vault X, and 3.21E-01 mrem/year for Vault Y.

### 8.1.4 Summary and Conclusion

The sensitivity analysis for Vault 4 has been considerably expanded from previous analysis to include key parameters and key radionuclides. All credible scenarios result in a dose to a member of the public less than 25 mrem/year. The results of the sensitivity analyses, when converted to dose, provide reasonable assurance that the Saltstone Disposal Facility will not exceed a dose to a member of the public of 25 mrem/year.

## 8.2 Vault 1 and 4 Plume Interaction Sensitivity

Vault 1 is an existing vault with waste located approximately 580 feet up gradient of Vault 4. There is potential plume interaction from Vault 1 based on stream traces (particle tracking). To quantify plume interaction from Vault 1, a conservative nonabsorbent tracer, nitrate, was chosen. Details of the sensitivity analysis for the plume interaction are in the 2005 SA, Section 7.5 (cook at al, 2005). A brief summary is provided below.

The nitrate SDF Vault 4 vadose zone model was used to compute the transient fractional release of nitrate from Vault 4 to the water table over the time period after IC to 10,000 years. The transient fractional release of nitrate from the SDF Vault 4 vadose zone was independently applied to both Vault 1 and Vault 4 aquifer source nodes. For each set of aquifer source nodes, the transient fractional release was partitioned to each aquifer source node by cell volume.

Two nitrate aquifer transport simulations were modeled: one case with only Vault 4 aquifer source nodes (base case), another with both Vault 1 and 4 aquifer source nodes active. The simulation time was from 0 to 10,000 years. The transient maximum concentration beyond the 100-ft point of assessment and a 100-m perimeter were determined.

### 8.2.1 Results of the Vault 1 and 4 Plume Interaction Sensitivity

The conclusion of the study is that there is no impact of plume interaction from Vault 1 for nitrate beyond the 100-ft point of assessment and the 1,000-year time of assessment. There appears to be an impact beyond the 100-m perimeter of Vault 4. However, the interaction only increases nitrate concentrations by about 25%. The Sum-of-Fractions of the 10,000-year groundwater limits is only 0.004. Applying a 25% reduction factor to all 10,000-year groundwater limits would only increase the Sum-of-Fractions to 0.005. Therefore, the results of the plume interaction sensitivity analyses, when converted to dose, provide reasonable assurance that the Saltstone Disposal Facility will not exceed a dose to a member of the public of 25 mrem/year.

## 8.3 Intruder Sensitivity

### 8.3.1 Post-Drilling Scenario as a Sensitivity Case

The concrete vault and saltstone are assumed to remain a barrier to drilling over the 10,000-year analysis period. However, to test the sensitivity of the results to this assumption, a post-drilling scenario is assessed. In this sensitivity analysis it is assumed that the reinforced concrete vault roof remains a barrier to excavation and drilling for a period of only 1,000 years. Therefore, the post-drilling sensitivity transient analysis was performed for times from 1,000 years to 10,000 years. Details of the sensitivity analysis for the post-drilling scenario as a sensitivity case are in the 2005 SA, Section 7.5 (Cook et al., 2005). A brief summary is provided below.

The post-drilling scenario assumes that an intruder who resides on the disposal site drills through a disposal unit in constructing a well for a domestic water supply. Following construction of the well, the contaminated material brought to the surface during drilling operations, which is assumed to be indistinguishable from native soil, is assumed to be mixed with native soil in the intruder's vegetable garden. In the post-drilling scenario, external and inhalation exposures, while residing in the home on the disposal site, are considered insignificant. All drilling waste is assumed to be mixed with native soil in the garden, which is considered to be at a sufficient distance from the home that indoor exposures are minor relative to those in the garden.

- ingestion of vegetables grown in the garden soil mixed with exhumed waste,
- direct ingestion of contaminated soil,
- external exposure to the contaminated soil while working in the garden, and
- inhalation of contaminated particulates while working in the garden.

#### 8.3.1.1 Results for Post-Drilling Scenario Sensitivity Analysis

Table 8-3 gives the results for the post-drilling scenario for the period 1,000 to 10,000 years. The entry “---“ in the Time of Limit column means that the dose calculation is always zero so there is no limit. For cases where there is a time given, there may be an entry “---“in one or both of the limit columns. In this case the entry “---“indicates a limit value greater than or equal to the threshold value of 1E+20 curies. (Cook et al., 2005)

The post-drilling limits are generally smaller (i.e., more restrictive) than the resident limits. If the post-drilling scenario were to be considered credible, the sum-of-fractions of the 10,000-year limits would increase from 0.21 to 0.31. While the sum of fractions increased in the sensitivity evaluation, the doses are still below the performance objective.

**Table 8-3. Intruder-Based Radionuclide Disposal Limits for Vault 4 Post-Drilling Scenario with Transient Calculation for 1,000 – 10,000 Years**

| Radionuclide | Time of Limit (Years) | Inventory Limit (Ci/Unit) |
|---|---|---|
| H-3 | 1000 | --- |
| C-14 | 1000 | 5.91E+03 |
| Na-22 | 1000 | --- |
| Al-26 | 1000 | 4.27E+03 |
| S-35 | --- | --- |
| Cl-36 | 1000 | 6.74E+01 |
| Ar-39 | 1000 | 9.75E+08 |
| K-40 | 1000 | 1.37E+03 |
| Ca-41 | 1000 | 3.21E+04 |
| Sc-46 | --- | --- |
| Co-60 | 1000 | --- |
| Ni-59 | 1000 | 1.12E+06 |
| Ni-63 | 1000 | 4.12E+08 |
| Se-79 | 1000 | 6.32E+04 |
| Kr-85 | 1000 | --- |
| Rb-87 | 1000 | 4.08E+04 |
| Sr-90 | 1000 | 1.14E+13 |
| Zr-93 | 1000 | 2.54E+06 |
| Nb-93m | 1000 | --- |
| Nb-94 | 1000 | 7.54E+03 |
| Mo-93 | 1000 | 1.48E+06 |
| Tc-99 | 1000 | 6.53E+03 |
| Pd-107 | 1000 | 2.33E+06 |
| Ag-108m | 1000 | 2.75E+04 |
| Cd-113m | 1000 | --- |
| Sn-121m | 1000 | 2.64E+11 |
| Sn-126 | 1000 | 5.57E+03 |
| Sb-125 | 1000 | --- |
| I-129 | 1000 | 1.01E+03 |
| Cs-134 | 1000 | --- |
| Cs-135 | 1000 | 6.51E+04 |
| Cs-137 | 1000 | 6.55E+13 |
| Ba-133 | 1000 | --- |
| Sm-151 | 1000 | 1.63E+10 |
| Eu-152 | 1000 | 4.83E+17 |
| Eu-154 | 1000 | --- |
| Eu-155 | 1000 | --- |
| W-181 | --- | --- |
| W-185 | --- | --- |
| W-188 | --- | --- |
| Pb-210 | 1000 | 8.01E+15 |
| Bi-207 | 1000 | 2.66E+13 |
| Ra-226 | 1000 | 2.75E+02 |

**Table 8-3. Intruder-Based Radionuclide Disposal Limits for Vault 4 Post-Drilling Scenario with Transient Calculation for 1,000 – 10,000 Years**

| Radionuclide | Time of Limit (Years) | Inventory Limit (Ci/Unit) |
|---|---|---|
| Ra-228 | 1000 | --- |
| Ac-227 | 1000 | 3.10E+16 |
| Th-228 | 1000 | --- |
| Th-229 | 1000 | 1.46E+03 |
| Th-230 | 9090 | 1.94E+02 |
| Th-232 | 1000 | 3.96E+02 |
| Pa-231 | 1000 | 3.32E+02 |
| U-232 | 1000 | 2.15E+07 |
| U-233 | 10000 | 1.83E+03 |
| U-234 | 10000 | 2.10E+03 |
| U-235 | 10000 | 1.45E+03 |
| U-236 | 1000 | 1.05E+04 |
| U-238 | 10000 | 9.95E+03 |
| Np-237 | 10000 | 2.91E+02 |
| Pu-238 | 10000 | 5.95E+06 |
| Pu-239 | 1000 | 4.04E+03 |
| Pu-240 | 1000 | 4.36E+03 |
| Pu-241 | 1000 | 4.58E+05 |
| Pu-242 | 1000 | 4.14E+03 |
| Pu-244 | 10000 | 2.30E+03 |
| Am-241 | 1000 | 1.56E+04 |
| Am-242m | 1000 | 1.70E+05 |
| Am-243 | 1000 | 3.26E+03 |
| Cm-242 | 10000 | 1.17E+09 |
| Cm-243 | 1000 | 3.32E+06 |
| Cm-244 | 1000 | 1.58E+06 |
| Cm-245 | 1600 | 1.98E+03 |
| Cm-246 | 1000 | 4.46E+03 |
| Cm-247 | 10000 | 2.04E+03 |
| Cm-248 | 1000 | 1.05E+03 |
| Bk-249 | 1000 | 5.92E+06 |
| Cf-249 | 1000 | 1.53E+04 |
| Cf-250 | 1000 | 1.62E+06 |
| Cf-251 | 1000 | 6.23E+03 |
| Cf-252 | 1000 | 1.43E+08 |

### 8.3.2 Agriculture Scenario Following Failure of Erosion Barrier Sensitivity

#### 8.3.2.1 Agriculture Scenario Following Failure of Erosion Barrier Sensitivity Description

In the inadvertent intruder analysis, the long-term persistence of the erosion barrier is assumed to preclude the Agricultural Scenario by maintaining a distance greater than that required to excavate a basement (10 ft.). To explore the sensitivity of the analysis results to this assumption, an alternate scenario in which the erosion barrier was assumed to erode at the same rate as the other cover material was assessed. Details of the sensitivity analysis for the agriculture scenario following failure of the erosion barrier are in the 2005 SA, Section 7.5. A brief summary is provided below.

For the agricultural scenario, the inadvertent intruder is assumed to be exposed to waste exhumed from the disposal unit while excavating to build a foundation for a home. In addition, the waste is assumed to be mixed with the native soil in a vegetable garden. Potential exposure pathways for the inadvertent intruder under this scenario include:

- ingestion of vegetables grown in the garden soil mixed with exhumed waste,
- direct ingestion of contaminated soil,
- external exposure to the contaminated soil while working in the garden,
- external exposure to the contaminated soil while residing in the home,
- inhalation of contaminated particulates while working in the garden, and
- inhalation of contaminated particulates while residing in the home.

#### 8.3.2.2. Agriculture Scenario Following Failure of Erosion Barrier Sensitivity Results

The erosion barrier is constructed of material sized to remain in place during a rainfall event with a 10,000-year recurrence interval calculated using an extreme-value distribution, i.e., 3.3 inches of rain in a 15 minute time span, (Weber 1998). Thus, the scenario is not credible. However, the dose for this sensitivity case was calculated and is 150 mrem/yr. This dose is still compliant with the 500 mrem to the whole body dose that has been chosen for compliance with 10 CFR 61.42 (USNRC, 1982).

**Table 8-4. Intruder-Based Radionuclide Disposal Limits for Vault 4 – Agriculture Scenario Following Failure of Erosion Barrier with Transient Calculation for 100 – 10,000 Years**

| Radionuclide | Time of Limit (Years) | Inventory Limit (Ci/Unit) |
|---|---|---|
| C-14 | 3275 | 1.30E+03 |
| Al-26 | 3275 | 3.44E+00 |
| Cl-36 | 3275 | 1.13E+01 |
| Ar-39 | 1132 | 1.29E+06 |
| K-40 | 3275 | 4.60E+01 |
| Ca-41 | 3275 | 5.44E+03 |
| Ni-59 | 3275 | 1.91E+05 |
| Ni-63 | 1280 | 6.87E+09 |
| Se-79 | 3275 | 1.05E+04 |
| Rb-87 | 3275 | 6.70E+03 |
| Sr-90 | 1132 | 1.66E+15 |
| Zr-93 | 3275 | 2.06E+05 |
| Nb-94 | 1132 | 6.45E+00 |
| Mo-93 | 1720 | 1.03E+05 |
| Tc-99 | 3275 | 1.09E+03 |
| Pd-107 | 3275 | 3.85E+05 |
| Ag-108m | 1132 | 4.07E+01 |
| Sn-121m | 1132 | 4.47E+10 |
| Sn-126 | 1132 | 5.11E+00 |
| I-129 | 3275 | 1.63E+02 |
| Cs-135 | 3275 | 1.08E+04 |
| Cs-137 | 1132 | 3.79E+12 |
| Sm-151 | 1132 | 7.50E+10 |
| Eu-152 | 3275 | 3.25E+16 |
| Pb-210 | 1150 | 9.56E+18 |
| Bi-207 | 1132 | 4.06E+11 |
| Ra-226 | 1132 | 8.76E+00 |
| Ac-227 | 1132 | 9.32E+16 |
| Th-229 | 1132 | 3.55E+01 |
| Th-230 | 9080 | 4.92E+00 |
| Th-232 | 3275 | 3.46E+00 |
| Pa-231 | 3275 | 1.48E+01 |
| U-232 | 1132 | 5.00E+05 |
| U-233 | 10000 | 4.49E+01 |
| U-234 | 10000 | 6.33E+01 |
| U-235 | 10000 | 3.66E+01 |
| U-236 | 3275 | 1.18E+03 |
| U-238 | 10000 | 3.16E+02 |
| Np-237 | 10000 | 2.47E+01 |
| Pu-238 | 10000 | 1.80E+05 |

**Table 8-4. Intruder-Based Radionuclide Disposal Limits for Vault 4 – Agriculture Scenario Following Failure of Erosion Barrier with Transient Calculation for 100 – 10,000 Years**

| Radionuclide | Time of Limit (Years) | Inventory Limit (Ci/Unit) |
|---|---|---|
| Pu-239 | 3275 | 4.45E+02 |
| Pu-240 | 3275 | 5.73E+02 |
| Pu-241 | 1132 | 1.05E+05 |
| Pu-242 | 3275 | 4.28E+02 |
| Pu-244 | 10000 | 2.65E+01 |
| Am-241 | 1132 | 3.58E+03 |
| Am-242m | 1132 | 6.56E+04 |
| Am-243 | 1132 | 7.00E+01 |
| Cm-242 | 10000 | 3.53E+07 |
| Cm-243 | 3275 | 3.53E+05 |
| Cm-244 | 3275 | 2.07E+05 |
| Cm-245 | 3275 | 1.08E+02 |
| Cm-246 | 3275 | 6.31E+02 |
| Cm-247 | 10000 | 2.24E+01 |
| Cm-248 | 3275 | 1.08E+02 |
| Bk-249 | 1132 | 1.05E+05 |
| Cf-249 | 1132 | 2.70E+02 |
| Cf-250 | 3275 | 2.29E+05 |
| Cf-251 | 1132 | 2.38E+02 |
| Cf-252 | 3275 | 1.47E+07 |

**Table 8-5. Comparison of 10,000-Year Agriculture Scenario Limits with Projected Inventory**

| Radionuclide | Limit, Ci | Estimated Inventory, Ci | Fraction of Limit |
|---|---|---|---|
| Am-241 | 3.58E+03 | 4.93E+02 | 1.38E-01 |
| Am-242m | 6.56E+04 | 3.31E+02 | 5.05E-03 |
| Am-243 | 7.00E+01 | 1.30E-03 | 1.86E-05 |
| C-14 | 1.30E+03 | 4.44E+00 | 3.43E-03 |
| Cf-251 | 2.38E+02 | 2.47E-01 | 1.04E-03 |
| Cm-243 | 3.53E+05 | 8.06E-02 | 2.28E-07 |
| Cm-244 | 2.07E+05 | 4.19E+02 | 2.02E-03 |
| Cm-245 | 1.08E+02 | 7.91E-02 | 7.33E-04 |
| Cs-135 | 1.08E+04 | 2.29E-02 | 2.12E-06 |
| Cs-137 | 3.79E+12 | 1.25E+06 | 3.29E-07 |
| Eu-152 | 3.25E+16 | 5.14E-03 | 1.58E-19 |
| I-129 | 1.63E+02 | 8.09E-01 | 4.96E-03 |
| Nb-94 | 6.45E+00 | 9.91E-04 | 1.54E-04 |
| Ni-59 | 1.91E+05 | 3.35E+00 | 1.75E-05 |
| Ni-63 | 6.87E+09 | 4.23E+00 | 6.15E-10 |
| Np-237 | 2.47E+01 | 7.23E-01 | 2.93E-02 |
| Pu-238 | 1.80E+05 | 3.33E+03 | 1.85E-02 |
| Pu-239 | 4.45E+02 | 4.20E+01 | 9.43E-02 |
| Pu-240 | 5.73E+02 | 7.74E+01 | 1.35E-01 |
| Pu-241 | 1.05E+05 | 1.55E+03 | 1.48E-02 |
| Pu-242 | 4.28E+02 | 1.56E-01 | 3.64E-04 |
| Se-79 | 1.05E+04 | 1.99E+00 | 1.89E-04 |
| Sm-151 | 7.50E+10 | 9.29E-04 | 1.24E-14 |
| Sn-126 | 5.11E+00 | 2.65E+00 | 5.19E-01 |
| Sr-90 | 1.66E+15 | 1.24E+05 | 7.47E-11 |
| Tc-99 | 1.09E+03 | 9.82E+01 | 8.98E-02 |
| Th-232 | 3.46E+00 | 3.62E-03 | 1.05E-03 |
| U-232 | 5.00E+05 | 9.46E+00 | 1.89E-05 |
| U-233 | 4.49E+01 | 1.46E+01 | 3.25E-01 |
| U-234 | 6.33E+01 | 6.53E+00 | 1.03E-01 |
| U-235 | 3.66E+01 | 7.91E-02 | 2.16E-03 |
| U-236 | 1.18E+03 | 1.85E-01 | 1.57E-04 |
| U-238 | 3.16E+02 | 3.61E-01 | 1.14E-03 |
| | | Sum-of-Fractions | 1.49E+00 |

### 8.3.3 Potential Water Usage Inside the 100 meter Buffer Zone Sensitivity

#### 8.3.3.1 Potential Water Usage Inside the 100 meter Buffer Zone Sensitivity Scenario Description

The intruder analyses in the 1992 PA and subsequent 2002 SA and 2005 SA did not include dose from use of contaminated groundwater. Additionally, the intruder analyses argued that the physical integrity of a Saltstone vault would prevent drilling through it for 10,000 years. To determine the dose to a hypothetical inadvertent intruder who is presumed to drill a well near, but not through, a Saltstone vault, and use the water for a variety of purposes (e.g., drinking, irrigating a garden), the following analysis was conducted.

The groundwater modeling in the Vault 4 SA (Cook et al., 2005) did not monitor groundwater concentrations at points nearer than 100 feet from a vault. Therefore, groundwater concentrations immediately under a vault were estimated by assuming that the maximum radionuclide flux leaving the vadose zone in a year was contained in the volume of water in the first layer of the model nodes in the saturated zone below the vault. This is conservative because the groundwater concentrations are from the water directly below the vaults, does not account for concentration dilution within the water table, and uses all of the activity released in a year in that volume of water. Figure 4.2 presents the upper portion of the model (i.e. the vadose zone). The flux to the water table is the amount of contaminant crossing into the water table indicated at the 0 foot elevation in Figure 4-2. These groundwater concentrations were used to calculate, for each radionuclide, the all-pathways dose from use of the water.

The peak radionuclide flux over 10,000 years was obtained from Table A-11 of Cook et al., 2005 (The tables and pertinent text from Cook et al., 2005 are reproduced in Appendix E.). The volume of the first layer of groundwater model nodes below Vault 4 is 1.73E7 L. Since the porosity of the soil is 0.42, the volume of water in the first layer of groundwater model nodes below Vault 4 is 7.27E6 L.

The radionuclide composition of salt waste for disposal in the Saltstone Disposal Facility has recently been revised (d'Entremont and Drumm, 2005). The revised projected inventory of radionuclides in Vault 4 is shown in Table 8-6.

Table 8-7 shows the peak fractional radionuclide flux from the vadose zone, the peak fractional radionuclide concentration, the revised projected inventory in Vault 4, and the estimated maximum concentration in groundwater under Vault 4 using the radionuclide inventory in Table 8-6.

**Table 8-6 Projected Vault 4 Radionuclide Inventory**

| Radionuclide | Curies | Radionuclide | Curies | Radionuclide | Curies |
|---|---|---|---|---|---|
| H-3 | 2.43E+03 | Cs-137 | 1.20E+06 | Np-237 | 5.76E-01 |
| C-14 | 6.88E+01 | Ba-137m | 1.13E+06 | Pu-238 | 3.69E+03 |
| Na-22 | 2.59E+02 | Ce-144 | 3.46E-01 | Pu-239 | 3.36E+01 |
| Al-26 | 1.03E+00 | Pr-144 | 3.46E-01 | Pu-240 | 8.39E+00 |
| Ni-59 | 3.46E-01 | Pm-147 | 2.93E+02 | Pu-241 | 1.72E+02 |
| Co-60 | 4.46E+01 | Sm-151 | 3.04E+02 | Pu-242 | 9.32E-03 |
| Ni-63 | 8.77E+01 | Eu-152 | 1.48E+00 | Am-241 | 1.44E+01 |
| Se-79 | 1.96E+00 | Eu-154 | 8.10E+01 | Am-242m | 7.52E-03 |
| Sr-90 | 5.29E+03 | Eu-155 | 1.72E+01 | Pu-244 | 9.38E-06 |
| Y-90 | 5.29E+03 | Ra-226 | 2.44E-01 | Am-243 | 6.22E-03 |
| Nb-94 | 1.02E-03 | Ra-228 | 6.41E-06 | Cm-242 | 6.21E-03 |
| Tc-99 | 7.16E+02 | Ac-227 | 1.37E-06 | Cm-243 | 2.88E-03 |
| Ru-106 | 4.82E+01 | Th-229 | 2.79E-03 | Cm-244 | 3.16E+00 |
| Rh-106 | 4.82E+01 | Th-230 | 1.49E-03 | Cm-245 | 3.03E-04 |
| Sb-125 | 2.05E+02 | Pa-231 | 3.80E-06 | Cm-247 | 5.55E-13 |
| Te-125m | 4.98E+01 | Th-232 | 6.41E-06 | Cm-248 | 5.79E-13 |
| Sn-126 | 9.56E+00 | U-232 | 9.52E-03 | Bk-249 | 4.23E-20 |
| Sb-126 | 1.33E+00 | U-233 | 9.82E-01 | Cf-249 | 3.21E-12 |
| Sb-126m | 9.50E+00 | U-234 | 6.59E+00 | Cf-251 | 2.47E-01 |
| I-129 | 4.40E-01 | U-235 | 7.41E-02 | Cf-252 | 3.56E-15 |
| Cs-134 | 2.40E+03 | U-236 | 1.42E-01 | | |
| Cs-135 | 4.14E+00 | U-238 | 1.61E-01 | | |

| Table 8-7 Estimated Peak Radionuclide Concentrations Below Saltstone Vault 4 | | | | | |
|---|---|---|---|---|---|
| Nuclide | Daughter | Peak Fractional Flux Ci/yr/Ci[a] | Peak Fractional Concentration pCi/L/Ci | Projected Inventory Ci/Vault 4 | Estimated Peak Concentration, pCi/L |
| Am-243 | | 1.43E-32 | 1.96E-27 | 6.22E-03 | 1.22E-29 |
| | Np-239 | 4.53E-36 | 6.22E-31 | | 3.87E-33 |
| | Pu-239 | 4.53E-27 | 6.22E-22 | | 3.87E-24 |
| | Pu-5-239 | 1.65E-30 | 2.27E-25 | | 1.41E-27 |
| C-14 | | 3.44E-24 | 4.73E-19 | 6.88E+01 | 3.25E-17 |
| Cm-245 | | 1.24E-38 | 1.70E-33 | 3.03E-04 | 5.15E-37 |
| | Pu-241 | 4.48E-40 | 6.15E-35 | | 1.86E-38 |
| | Pu5-241 | 1.75E-43 | 2.40E-38 | | 7.27E-42 |
| | Am-241 | 2.32E-37 | 3.19E-32 | | 9.67E-36 |
| | Np-237 | 3.96E-24 | 5.44E-19 | | 1.65E-22 |
| Cs-135 | | 1.10E-14 | 1.51E-09 | 4.14E+00 | 6.25E-09 |
| Cs-137 | | 1.42E-41 | 1.95E-36 | 1.20E+06 | 2.34E-30 |
| H-3 | | 4.03E-13 | 5.54E-08 | 2.43E+03 | 1.35E-04 |
| I-129 | | 1.29E-07 | 1.77E-02 | 4.40E-01 | 7.79E-03 |
| Nb-94 | | 3.33E-21 | 4.57E-16 | 1.02E-03 | 4.67E-19 |
| Ni-59 | | 2.37E-18 | 3.26E-13 | 3.46E-01 | 1.13E-13 |
| Np-237 | | 7.25E-24 | 9.96E-19 | 5.76E-01 | 5.74E-19 |
| Pu-238 | | 5.59E-42 | 7.68E-37 | 3.69E+03 | 2.83E-33 |
| | Pu5-238 | 2.07E-45 | 2.84E-40 | | 1.05E-36 |
| | U-234 | 4.13E-26 | 5.67E-21 | | 2.09E-17 |
| Pu-239 | | 7.75E-27 | 1.06E-21 | 3.36E+01 | 3.56E-20 |
| | Pu5-239 | 2.81E-30 | 3.86E-25 | | 1.30E-23 |
| | U-235 | 1.83E-27 | 2.51E-22 | | 8.43E-21 |
| Pu-240 | | 3.59E-27 | 4.93E-22 | 8.39E+00 | 4.14E-21 |
| | Pu5-240 | 1.30E-30 | 1.79E-25 | | 1.50E-24 |
| | U-236 | 5.85E-27 | 8.04E-22 | | 6.75E-21 |
| Pu-241 | | 3.93E-68 | 5.40E-63 | 1.72E+02 | 9.28E-61 |
| | Pu5-241 | 1.64E-71 | 2.25E-66 | | 3.87E-64 |
| | Am-241 | 4.00E-39 | 5.49E-34 | | 9.45E-32 |
| | Np-237 | 7.25E-24 | 9.96E-19 | | 1.71E-16 |
| Pu-242 | | 1.01E-26 | 1.39E-21 | 9.32E-03 | 1.29E-23 |
| | Pu5-242 | 3.68E-30 | 5.05E-25 | | 4.71E-27 |
| | U-238 | 1.26E-28 | 1.73E-23 | | 1.61E-25 |
| Se-79 | | 7.11E-07 | 9.77E-02 | 1.96E+00 | 1.91E-01 |
| Sn-126 | | 2.03E-22 | 2.79E-17 | 9.56E+00 | 2.67E-16 |
| Sr-90 | | 4.32E-19 | 5.93E-14 | 5.29E+03 | 3.14E-10 |
| Tc-99 | | 5.61E-20 | 7.71E-15 | 7.16E+02 | 5.52E-12 |
| Th-232 | | 3.13E-36 | 4.30E-31 | 6.41E-06 | 2.76E-36 |
| | Ra-228 | 9.13E-45 | 1.25E-39 | | 8.04E-45 |
| | Th-228 | 4.74E-46 | 6.51E-41 | | 4.17E-46 |

| Table 8-7 Estimated Peak Radionuclide Concentrations Below Saltstone Vault 4 | | | | | |
|---|---|---|---|---|---|
| Nuclide | Daughter | Peak Fractional Flux Ci/yr/Ci[a] | Peak Fractional Concentration pCi/L/Ci | Projected Inventory Ci/Vault 4 | Estimated Peak Concentration, pCi/L |
| | Ra-224 | 1.59E-47 | 2.18E-42 | | 1.40E-47 |
| U-232 | | 2.38E-48 | 3.27E-43 | 9.52E-03 | 3.11E-45 |
| | Th-228 | 1.66E-50 | 2.28E-45 | | 2.17E-47 |
| | Ra-224 | 5.58E-52 | 7.66E-47 | | 7.30E-49 |
| U-233 | | 4.45E-26 | 6.11E-21 | 9.82E-01 | 6.00E-21 |
| | Th-229 | 5.04E-29 | 6.92E-24 | | 6.80E-24 |
| | Ra-225 | 1.79E-33 | 2.46E-28 | | 2.42E-28 |
| U-234 | | 4.52E-26 | 6.21E-21 | 6.59E+00 | 4.09E-20 |
| | Th-230 | 3.58E-29 | 4.92E-24 | | 3.24E-23 |
| | Ra-226 | 2.86E-23 | 3.93E-18 | | 2.59E-17 |
| | Pb-210 | 7.72E-25 | 1.06E-19 | | 6.99E-19 |
| | Po-210 | 2.36E-26 | 3.24E-21 | | 2.14E-20 |
| U-235 | | 4.65E-26 | 6.39E-21 | 7.41E-02 | 4.73E-22 |
| | Pa-321 | 1.09E-30 | 1.50E-25 | | 1.11E-26 |
| | Ac-227 | 8.86E-34 | 1.22E-28 | | 9.02E-30 |
| | Th-227 | 2.93E-37 | 4.02E-32 | | 2.98E-33 |
| | Ra-223 | 1.15E-36 | 1.58E-31 | | 1.17E-32 |
| U-236 | | 4.65E-26 | 6.39E-21 | 1.42E-01 | 9.07E-22 |
| U-238 | | 4.65E-26 | 6.39E-21 | 1.61E-01 | 1.03E-21 |
| | Th-234 | 1.72E-37 | 2.36E-32 | | 3.80E-33 |
| | U-234 | 7.12E-32 | 9.78E-27 | | 1.57E-27 |

#### 8.3.3.2 Results From Potential Water Usage Inside the 100 meter Buffer Zone Sensitivity

The dose from all-exposure pathways (e.g., drinking water, eating crops irrigated by groundwater) from the use of groundwater under Saltstone Vault 4 is shown in Table 8-8. The dose was calculated from the peak groundwater concentrations using the LADTAP XL program (Simpkins 2004b), which is an SRS implementation of the NRC code. The total dose is calculated to be 0.27 mrem/year. This total dose is very conservative in that it assumes that the peak groundwater concentrations for each radionuclide are coincident in time.

**Table 8-8. Peak All-Pathways Dose from Use of Groundwater Below Saltstone Vault 4**

| Nuclide | Peak All-Pathways Dose, mrem/year |
|---|---|
| H-3 | 1.17E-08 |
| C-14 | 3.45E-17 |
| Ni-59 | 1.50E-16 |
| Se-79 | 2.56E-01 |
| Sr-90 | 1.66E-10 |
| Nb-94 | 1.43E-18 |
| Tc-99 | 6.44E-13 |
| Sn-126 | 2.86E-16 |
| I-129 | 1.05E-02 |
| Cs-135 | 2.73E-09 |
| Cs-137 | 7.24E-30 |
| Th-232 | 2.78E-35 |
| U-232 | 2.84E-45 |
| U-233 | 5.02E-21 |
| U-234 | 1.92E-16 |
| U-235 | 3.65E-22 |
| U-236 | 6.90E-22 |
| U-238 | 7.21E-22 |
| Np-237 | 7.04E-18 |
| Pu-238 | 1.66E-17 |
| Pu-239 | 4.54E-19 |
| Pu-240 | 5.72E-20 |
| Pu-241 | 2.10E-15 |
| Pu-242 | 1.55E-22 |
| Am-243 | 4.88E-23 |
| Cm-245 | 2.02E-21 |
| Total | 2.67E-01 |

## 8.4 Impact of Cover and Vault Degradation Beyond 10,000 Years

Details of the sensitivity analysis for the impact of cover and vault degradation beyond 10,000 years are in 2005 SA, Section 7.5 (Cook et al., 2005). A brief summary is provided below.

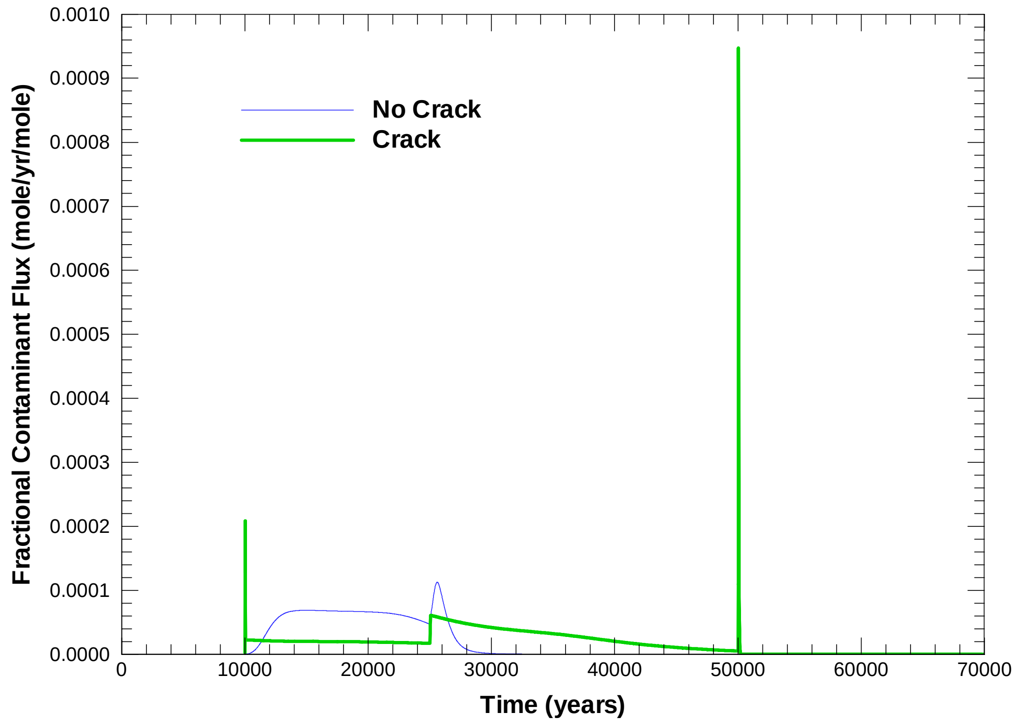
From 10,000 to about 12,000 years, the gravel drainage layer overlying the vault roof is predicted to completely silt up with fines (Phifer 2004b), producing a significantly lower hydraulic conductivity. The lower hydraulic conductivity estimate is conservatively assumed to apply over the entire 10,000 to 25,000 year period in model simulations. Without macroscopic cracks in Saltstone, water ponds over the vault roof from 10,000 to 50,000 years in PORFLOW flow simulations. The increased hydraulic head gradient driving flow through Saltstone, coupled with moderately increased Saltstone and concrete conductivities compared to earlier times, produces a higher fractional flux due to post-10,000 year degradation. Flux peaks occur shortly after 10,000 and 25,000 years in response to step changes in the modeled properties for Saltstone and concrete.

However, under ponded water or positive pressure conditions, large-scale cracks are expected to preferentially transmit water compared to the surrounding matrix. The additional effect of cracks on flow and water table flux was considered in a second sensitivity run. The physical cracks are predicted to occur at a 30 ft spacing within the plane of the two dimensional PORFLOW vadose zone model, which is a typical cross-section of the long axis of the vault. The presence of cracks in the model prevents water from ponding on the vault roof, but provides sudden pathways for water to infiltrate the core of the Saltstone waste. The resulting flux transient for I-129 is shown in Figure 8-1. A very sharp peak in flux is observed immediately following 10,000 years, when the cracks suddenly become active in the simulations. The flux is diffusion-limited, and stabilizes to a much lower value after I-129 is leached from Saltstone near the crack faces. A second peak occurs at 25,000 years in response to increased Saltstone conductivity, similar to the no-crack sensitivity run. At 50,000 years, the conductivity of Saltstone is assumed to increase by two orders of magnitude, and the remaining inventory flushes from the vault by advection.

To a large extent, the abrupt changes in flux observed in the simulations including cracks are an artifact of simulating transport using a sequence of steady-state flow fields. In reality, the flow conditions would change gradually over time, and the flux transient would be much smoother than depicted in Figure 8-1. In particular, flux peaks are expected to be lower in peak magnitude, but broader in duration.

This study demonstrates the importance of the drainage layer at the top of the vault. The time over which the layer continues to function could be increased by making the layer thicker but it is not necessary for the 10,000 year compliance period.

**Figure 8-1. Instantaneous I-129 Fractional Contaminant Flux to the Water Table (10,000 to 70,000 yrs) Assuming Cover and Vault Degradation, With and Without Cracks.**



## 8.5 Uncertainty

Details of the uncertainty analysis are given in the 2005 SA, Section 7.5. A brief summary is provided below.

The projected impacts from Saltstone disposal are very low. The all-pathways inventory limits for SDF Vault 4 presented in Table 4-18 are very large in comparison with the projected inventory shown in Table 3-2. The sum-of-fractions for the groundwater pathway is 4.6E-08, the sum-of-fractions for the air pathway is 5.2E-07, and that for all pathways is 2.4E-07.

It is clear from the very low sums-of-fractions that the calculated disposal limits would have to decrease by several orders of magnitude for the impacts from Saltstone disposal in the SDF to approach an appreciable fraction of one of the performance objectives. For example, in the all pathways analysis, the sum-of-fractions of the limits is $2.4 \times 10^{-7}$. If each of these disposal limits decreased by four orders of magnitude (i.e., by a factor of 10,000), the sum-of-fractions would still be only 0.0024, which would represent a dose of only 0.06 mrem/year.

In the 1992 PA, analyses of the sensitivity of model results to parameter changes and of uncertainty were performed for the groundwater pathway. These analyses investigated the fluxes to the water table from intact and degraded vaults. The most sensitive parameters for intact vaults were: 1) the saturated hydraulic conductivity of Saltstone, 2) the diffusivity of nitrate in the Saltstone, 3) the saturated hydraulic conductivity of concrete; and 4) the diffusivity of nitrate in the concrete. For degraded vaults, the sensitivity analysis considered depth of perched water on top of the vaults, crack spacing, crack aperture, and distribution coefficient. A sensitivity analysis for the groundwater flow and transport model used in the 1992 PA was also conducted. The most sensitive parameter was the saturated hydraulic conductivity of the hydrologic units considered in the model. Latin Hypercube Sampling techniques were applied to the parameter distributions studied to estimate uncertainty in the nitrate concentration in the groundwater at the 100-meter point of assessment.

In evaluating uncertainties in doses to future members of the public, projected over long time-frames, the most important consideration may be the definitions of the exposure scenarios. In the 2005 SA, as in the 1992 PA, it was assumed that a future member of the public would have access to the land within 100 meters of the disposed waste. However, the SRS Land Use Plan (USDOE, 2000) requires Federal ownership and control of the site well beyond 100 years after closure of SDF. DOE 5400.5 precludes release of the

area unless the radiological hazard meets the requirements of DOE 5400.5 Chapter 4, which essentially requires perpetual DOE control. No unrestricted use of the land or groundwater will be permitted for the central portion of the site, which includes SDF. Thus, a member of the public could not contact the groundwater in the vicinity of SDF. Due to the restrictions in the SRS Land Use Plan, a member of the public could only contact potentially contaminated surface water off-site, approximately six miles from the facility at the mouth of Upper Three Runs. Furthermore, concentrations of SDF radionuclides in that surface water will be much less than that assessed in the SA at 100 meters from SDF due to decay and other natural processes.

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## 9.0 Conclusion

This PODD addresses the disposal of solidified low-activity salt waste streams into the SDF as saltstone grout and its compliance with performance objectives for near-surface disposal of radioactive waste. Specifically, this PODD demonstrates and documents that the solidified low-activity salt streams from the SRS salt processing activities meet the performance objectives set out in Subpart C of Part 61 of Title 10, Code of Federal Regulations. The performance objectives from 10 CFR 61 and the results demonstrating compliance are as follows:

- 10 CFR 61.41 – The all-pathways dose-based performance objective is that no member of the public may receive an annual dose exceeding 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ. The evaluation of this performance objective is presented in Section 4. The calculated annual doses to a member of the public are 2.3 mrem to the whole body, 4.6 mrem to the thyroid and 5.3 mrem to any other organ. These doses are well within the 10 CFR 61.41 performance objective.

- 10 CFR 61.42 – The inadvertent intruder dose-based performance objective is to protect any member of the public from intruder scenarios. The 500 mrem to the whole body has been chosen from the basis for 10 CFR 61 waste classification discussed in Section 5.2 of the Final Environmental Impact Statement on 10 CFR 61 (USNRC, 1982). The evaluation of this performance objective is presented in Section 5. The calculated annual dose to the inadvertent intruder of 22 mrem to the whole body is well within the 10 CFR 61.42 performance objective.

- 10 CFR 61.43 – The operational radiation exposure performance objective is that operation of the SDF is conducted in compliance with 10 CFR 20 (USNRC, 2005) and 10CFR 61.41. (10 CFR 61.41 is addressed above) The evaluation of this performance objective is presented in Section 6. The occupational radiation protection program ensures compliance with the 10 CFR 61.43 performance objective.

- 10 CFR 61.44 – The disposal facility performance objective is that the design, operation and closure of the SDF will achieve long-term stability of the site to eliminate to the extent practicable the need for ongoing maintenance. The evaluation of this performance objective is presented in Section 7. The disposal facility design ensures compliance with the 10 CFR 61.44 performance objective.

Based upon the information presented in this PODD, the SRS low-activity salt stream to be disposed of in the SDF complies with all 10 CFR 61 Subpart C performance objectives for the disposal of SRS salt waste streams.

## 10.0 References

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# APPENDIX A.  Interim Salt Strategy Technology Description

**Deliquification Dissolution and Adjustment (DDA), Actinide Removal Process (ARP) and Modular Caustic side solvent extraction Unit (MCU)**

DOE identified three technologies for removing radionuclides from the salt waste that it could deploy during the interim period between now and approximately 2009 when SWPF is slated to come online.  These are DDA, ARP and MCU.

These processes were selected for their ability to reduce Cs-137, Sr-90, and actinide concentrations.  DOE assessed projected tank space availability versus needs during the time between now and the start-up of SWPF and determined that insufficient space existed for sustained sludge disposition activities or for feed preparation for SWPF. Options for nearer term deployment of treatment technologies were evaluated and, based upon the technology selection process used for SWPF, DOE decided to pursue early small-scale implementation of the same actinide and Cs removal technologies (i.e., ARP and MCU).  Capacity limitations imposed by the re-use of existing facilities necessitated initiation of salt removal prior to availability of the small-scale facilities.  DDA was selected as the best alternative to augment ARP/MCU capacities until SWPF was operational.

DOE anticipates using all three technologies in combination in approximately 2007, when ARP and MCU are expected to come online.  In the meantime, upon issuance of a final 3116 Determination **(***Ronald W. Reagan National Defense Authorization Act for FY 2005*, Section 3116, 2004.) DOE anticipates using DDA alone to process some of the lowest activity salt waste.

**DDA**

The deliquification and dissolution processes involved in DDA will remove substantial amounts of Cs-137, as well as some insoluble Sr and actinides, from this already relatively low-curie material[6]. Through deliquification, DDA will remove approximately 50% (Hopkins and Shah, 2004) of the Cs-137 and its daughter product Ba-137m from the saltcake targeted for dissolution.  The liquid removed from the saltcake will be transferred to other tanks for future processing by SWPF.  The saltcake will then be dissolved, after which it will be managed in a manner that will facilitate the settling of the insoluble sludge solids (e.g., Sr-90 and actinides) entrained in the salt solution and minimize the transfer of these materials to SDF.  This is done by transferring the dissolved salt solution to a staging tank where the solids are allowed to settle before transfer to the Saltstone Facility feed tank.  After being processed through DDA and after

[6] This process will not be used for Tank 48 waste.

being solidified in a grout matrix, the solidified low-activity salt waste will have concentrations below Class C limits for all radionuclides. DOE can begin to use DDA immediately.

**ARP and MCU**

ARP and MCU are expected to come online in approximately 2007. ARP and MCU will remove an additional approximately 92% (Campbell, 2004) of the Cs-137/Ba-137m following DDA while also removing insoluble solids which contain the majority of the Sr and actinides. The ARP facilities will also have the capability to remove soluble Sr and actinides through MonoSodium Titanate (MST) strikes. If the soluble actinides in the original salt solution are sufficiently low (i.e., below Class C concentration limits), to achieve the necessary tank space recovery prior to SWPF start-up, the stream will only be filtered prior to being sent to MCU. The filtering step will remove the majority of insoluble solids and is a necessary precursor to processing the salt solution through MCU[7]. Additionally, while neither ARP nor MCU preferentially targets the removal of Se-79, Tc-99, Sn-126, and I-129, ARP's filtration process will remove a majority of the insoluble species of these radionuclides.

**Use of ARP, MCU, and DDA**

Once ARP and MCU are online, in approximately 2007, DOE anticipates using them on waste that has already undergone DDA to process approximately 2.1 Mgal of salt waste that are expected to result in an estimated 0.3 MCi disposed of in SDF vaults from these facilities (Mahoney and Chew, 2004). By processing this salt solution through ARP and MCU, rather than just DDA, the total number of curies ultimately disposed in SDF vaults will be decreased by an estimated 3.1 MCi.

DOE anticipates using DDA alone to process approximately 6.0 Mgal of some of the lowest curie salt solution, resulting in an estimated 1.7 MCi disposed in SDF vaults from these streams (Mahoney and Chew, 2004). This is because if salt processing is delayed until the ARP/MCU facilities become operational in approximately 2007, this still will result in significant delay to DOE's sludge removal efforts. This, in turn, is because DOE has an immediate need for usable working space in specific new-style tanks within the Tank Farms to continue these activities. This particular space is needed because of the proximity of these tanks to the salt processing facilities and the existing transfer line infrastructure, as well as the need to have viable concentrate receipt tanks for the evaporator systems associated with sludge batch preparation and concentration of the recycle waste stream returning from DWPF. Without use of DDA to remove salt waste

[7] The current Interim Salt Processing Strategy does not generally contemplate MST strikes of the salt solutions that will be batched through ARP/MCU but an 8-hour MST strike will be performed if necessary to meet Class C limits for disposal in SDF or if throughputs can be maintained at 1.5 Mgal per year even if strikes are not necessary to meet Class C concentration limits.

until ARP and MCU are online, there will still be insufficient space created in the new-style tanks to ensure continued risk-reduction at DWPF at its current rate of processing and sufficient feed batches cannot be prepared to ensure that SWPF can start-up operations at the rate of 5 Mgal during the first twelve months of operation. In addition, the increased waste inventory will result in increased safety risks associated with the operational activities of the Tank Farms as identified by the DNFSB (Conway and McSlarrow, 2004).

After ARP/MCU are on-line, DOE will continue to use DDA alone in the interim period for some of the salt waste. This is because the capacity of the combined ARP/MCU facilities is only 1.5 Mgal salt solution per year and these facilities, therefore, cannot be used to process enough salt waste to create the tank space needed.

**Tank 48 Waste**

In addition to the low-activity salt wastes that DOE proposes to process during the interim period using combinations of DDA, ARP, and MCU, DOE also proposes to create additional tank space during this interim period by removing a unique, low-activity salt waste in Tank 48 (e.g. high organic content), aggregating it with DWPF recycle, and disposing of the aggregated salt waste stream in SDF as a solidified low-activity salt waste.

# APPENDIX B. Inadvertent Intruder Input Variables

**Table B-1. Radionuclides Considered in Dose Analysis for Inadvertent Intruders (Lee, 2004)**

| Nuclide | Half Life[1] | Units | Daughter1 | Branch1 | Daughter2 | Branch2 |
|---|---|---|---|---|---|---|
| Ac-225 | 1.0000E+01 | days | Fr-221 | 1 | | |
| Ac-227 | 2.1773E+01 | years | Fr-223 | 0.0138 | Th-227 | 0.9856 |
| Ac-228 | 6.1500E+00 | hours | Th-228 | 1 | | |
| Ag-108 | 2.3700E+00 | minutes | | | | |
| Ag-108m | 4.1800E+02 | years | Ag-108 | 0.089 | | |
| Al-26 | 7.1700E+05 | years | | | | |
| Am-241 | 4.3220E+02 | years | Np-237 | 1 | | |
| Am-242 | 1.6020E+01 | hours | Pu-242 | 0.173 | Cm-242 | 0.827 |
| Am-242m | 1.4100E+02 | years | Np-238 | 0.00476 | Am-242 | 0.995 |
| Am-243 | 7.3700E+03 | years | Np-239 | 1 | | |
| Ar-39 | 2.6900E+02 | years | | | | |
| At-217 | 3.2300E-02 | seconds | Bi-213 | 1 | | |
| At-218 | 1.5000E+00 | seconds | Bi-214 | 1 | | |
| Ba-133 | 3.8489E+03 | days | | | | |
| Ba-137m | 2.5520E+00 | minutes | | | | |
| Bi-207 | 3.1550E+01 | years | | | | |
| Bi-210 | 5.0130E+00 | days | Po-210 | 1 | | |
| Bi-211 | 2.1400E+00 | minutes | Tl-207 | 0.9972 | Po-211 | 0.0028 |
| Bi-212 | 6.0550E+01 | minutes | Tl-208 | 0.3593 | Po-212 | 0.6407 |
| Bi-213 | 4.5590E+01 | minutes | Tl-209 | 0.0216 | Po-213 | 0.9784 |
| Bi-214 | 1.9900E+01 | minutes | Po-214 | 0.9998 | | |
| Bk-249 | 3.3000E+02 | days | Cf-249 | 1 | | |
| C-14 | 5.7300E+03 | years | | | | |
| Ca-41 | 1.0300E+05 | years | | | | |
| Cd-113m | 1.4100E+01 | years | | | | |
| Cf-249 | 3.5100E+02 | years | Cm-245 | 1 | | |
| Cf-250 | 1.3080E+01 | years | Cm-246 | 0.9992 | | |
| Cf-251 | 8.9800E+02 | years | Cm-247 | 1 | | |
| Cf-252 | 2.6450E+00 | years | Cm-248 | 0.9691 | | |
| Cl-36 | 3.0100E+05 | years | | | | |
| Cm-242 | 1.6280E+02 | days | Pu-238 | 1 | | |
| Cm-243 | 2.9100E+01 | years | Pu-239 | 0.9976 | Am-243 | 0.0024 |
| Cm-244 | 1.8100E+01 | years | Pu-240 | 1 | | |
| Cm-245 | 8.5000E+03 | years | Pu-241 | 1 | | |
| Cm-246 | 4.7600E+03 | years | Pu-242 | 0.9997 | | |
| Cm-247 | 1.5600E+07 | years | Pu-243 | 1 | | |
| Cm-248 | 3.4800E+05 | years | Pu-244 | 0.9174 | | |

**Table B-1. Radionuclides Considered in Dose Analysis for Inadvertent Intruders (Lee, 2004)**

| Nuclide | Half Life[1] | Units | Daughter1 | Branch1 | Daughter2 | Branch2 |
|---|---|---|---|---|---|---|
| Co-60 | 1.9251E+03 | days | | | | |
| Cs-134 | 7.5450E+02 | days | | | | |
| Cs-135 | 2.3000E+06 | years | | | | |
| Cs-137 | 3.0070E+01 | years | Ba-137m | 0.946 | | |
| Eu-152 | 1.3516E+01 | years | Gd-152 | 0.278 | | |
| Eu-154 | 8.5920E+00 | years | | | | |
| Eu-155 | 4.7611E+00 | years | | | | |
| Fr-221 | 4.9000E+00 | minutes | At-217 | 1 | | |
| Fr-223 | 2.2000E+01 | minutes | Ra-223 | 1 | | |
| Gd-152 | 1.0800E+14 | years | | | | |
| H-3 | 1.2330E+01 | years | | | | |
| I-129 | 1.5700E+07 | years | | | | |
| K-40 | 1.2770E+09 | years | | | | |
| Kr-85 | 3.9344E+03 | days | | | | |
| Mo-93 | 4.0000E+03 | years | Nb-93m | 1 | | |
| Na-22 | 2.6019E+00 | years | | | | |
| Nb-93m | 1.6130E+01 | years | | | | |
| Nb-94 | 2.0300E+04 | years | | | | |
| Ni-59 | 7.6000E+04 | years | | | | |
| Ni-63 | 1.0010E+02 | years | | | | |
| Np-237 | 2.1440E+06 | years | Pa-233 | 1 | | |
| Np-238 | 2.1170E+00 | days | Pu-238 | 1 | | |
| Np-239 | 2.3565E+00 | days | Pu-239 | 1 | | |
| Np-240 | 6.1900E+01 | minutes | Pu-240 | 1 | | |
| Np-240m | 7.2200E+00 | minutes | Pu-240 | 1 | | |
| Pa-231 | 3.2760E+04 | years | Ac-227 | 1 | | |
| Pa-233 | 2.6967E+01 | days | U-233 | 1 | | |
| Pa-234 | 6.7000E+00 | hours | U-234 | 1 | | |
| Pa-234m | 1.1700E+00 | minutes | Pa-234 | 0.0013 | U-234 | 0.9987 |
| Pb-209 | 3.2530E+00 | hours | | | | |
| Pb-210 | 2.2300E+01 | years | Bi-210 | 1 | | |
| Pb-211 | 3.6100E+01 | minutes | Bi-211 | 1 | | |
| Pb-212 | 1.0640E+01 | hours | Bi-212 | 1 | | |
| Pb-214 | 2.6800E+01 | minutes | Bi-214 | 1 | | |
| Pd-107 | 6.5000E+06 | years | | | | |
| Po-210 | 1.3838E+02 | days | | | | |
| Po-211 | 5.1600E-01 | seconds | | | | |
| Po-212 | 2.9800E-07 | seconds | | | | |
| Po-213 | 3.6500E-06 | seconds | Pb-209 | 1 | | |
| Po-214 | 1.6430E-04 | seconds | Pb-210 | 1 | | |
| Po-215 | 1.7810E-03 | seconds | Pb-211 | 1 | | |

**Table B-1. Radionuclides Considered in Dose Analysis for Inadvertent Intruders (Lee, 2004)**

| Nuclide | Half Life[1] | Units | Daughter1 | Branch1 | Daughter2 | Branch2 |
|---|---|---|---|---|---|---|
| Po-216 | 1.4500E-01 | seconds | Pb-212 | 1 | | |
| Po-218 | 3.1000E+00 | minutes | Pb-214 | 0.9998 | At-218 | 0.0002 |
| Pu-238 | 8.7700E+01 | years | U-234 | 1 | | |
| Pu-239 | 2.4110E+04 | years | U-235 | 1 | | |
| Pu-240 | 6.5640E+03 | years | U-236 | 1 | | |
| Pu-241 | 1.4290E+01 | years | Am-241 | 1 | | |
| Pu-242 | 3.7330E+05 | years | U-238 | 1 | | |
| Pu-243 | 4.9560E+00 | hours | Am-243 | 1 | | |
| Pu-244 | 8.0000E+07 | years | U-240 | 0.9988 | | |
| Ra-223 | 1.1435E+01 | days | Rn-219 | 1 | | |
| Ra-224 | 3.6600E+00 | days | Rn-220 | 1 | | |
| Ra-225 | 1.4900E+01 | days | Ac-225 | 1 | | |
| Ra-226 | 1.6000E+03 | years | Rn-222 | 1 | | |
| Ra-228 | 5.7500E+00 | years | Ac-228 | 1 | | |
| Rb-87 | 4.7500E+10 | years | | | | |
| Re-188 | 1.7005E+01 | hours | | | | |
| Rn-219 | 3.9600E+00 | seconds | Po-215 | 1 | | |
| Rn-220 | 5.5600E+01 | seconds | Po-216 | 1 | | |
| Rn-222 | 3.8235E+00 | days | Po-218 | 1 | | |
| S-35 | 8.7380E+01 | days | | | | |
| Sb-125 | 2.7586E+00 | years | Te-125m | 0.228 | | |
| Sb-126 | 1.2460E+01 | days | | | | |
| Sb-126m | 1.9150E+01 | minutes | Sb-126 | 0.14 | | |
| Sc-46 | 8.3790E+01 | days | | | | |
| Se-79 | 1.1000E+06 | years | | | | |
| Sm-151 | 9.0000E+01 | years | | | | |
| Sn-121 | 2.7060E+01 | hours | | | | |
| Sn-121m | 5.5000E+01 | years | Sn-121 | 0.76 | | |
| Sn-126 | 1.0000E+05 | years | Sb-126m | 1 | | |
| Sr-90 | 2.8790E+01 | years | Y-90 | 1 | | |
| Tc-99 | 2.1110E+05 | years | | | | |
| Te-125m | 5.7400E+01 | days | | | | |
| Th-227 | 1.8720E+01 | days | Ra-223 | 1 | | |
| Th-228 | 1.9116E+00 | years | Ra-224 | 1 | | |
| Th-229 | 7.3400E+03 | years | Ra-225 | 1 | | |
| Th-230 | 7.5380E+04 | years | Ra-226 | 1 | | |
| Th-231 | 2.5520E+01 | hours | Pa-231 | 1 | | |
| Th-232 | 1.4050E+10 | years | Ra-228 | 1 | | |
| Th-234 | 2.4100E+01 | days | Pa-234m | 1 | | |
| Tl-207 | 4.7700E+00 | minutes | | | | |
| Tl-208 | 3.0530E+00 | minutes | | | | |

**Table B-1. Radionuclides Considered in Dose Analysis for Inadvertent Intruders (Lee, 2004)**

| Nuclide | Half Life[1] | Units | Daughter1 | Branch1 | Daughter2 | Branch2 |
|---|---|---|---|---|---|---|
| Tl-209 | 2.2000E+00 | minutes | Pb-209 | 1 | | |
| U-232 | 6.8900E+01 | years | Th-228 | 1 | | |
| U-233 | 1.5920E+05 | years | Th-229 | 1 | | |
| U-234 | 2.4550E+05 | years | Th-230 | 1 | | |
| U-235 | 7.0380E+08 | years | Th-231 | 1 | | |
| U-236 | 2.3420E+07 | years | Th-232 | 1 | | |
| U-238 | 4.4680E+09 | years | Th-234 | 1 | | |
| U-240 | 1.4100E+01 | hours | Np-240m | 1 | | |
| W-181 | 1.2120E+02 | days | | | | |
| W-185 | 7.5100E+01 | days | | | | |
| W-188 | 6.9400E+01 | days | Re-188 | 1 | | |
| Y-90 | 6.4000E+01 | hours | | | | |
| Zr-93 | 1.5300E+06 | years | Nb-93m | 1 | | |

NOTES:
[1] Tuli 2000

Taken from EPA 1988 and 1993.

**Table B-2. Intruder Input – Internal and External Dose Conversion Factors (Lee 2004)**

| | Internal DCFs (rem/μCi) | | External DCFs (rem/yr per μCi/m$^3$) | |
|---|---|---|---|---|
| **Nuclide** | **Ingestion** | **Inhalation** | **Infinite Depth** | **15 cm** |
| Ac-225 | 1.11E-01 | 1.08E+01 | 3.98E-05 | 3.90E-05 |
| Ac-227 | 1.41E+01 | 6.70E+03 | 3.10E-07 | 3.06E-07 |
| Ac-228 | 2.16E-03 | 3.08E-01 | 3.74E-03 | 3.22E-03 |
| Al-26 | 1.46E-02 | 7.96E-02 | 1.09E-02 | 9.03E-03 |
| Am-241 | 3.64E+00 | 4.44E+02 | 2.73E-05 | 2.73E-05 |
| Am-242 | 1.41E-03 | 5.85E-02 | 3.12E-05 | 3.12E-05 |
| Am-242m | 3.52E+00 | 4.26E+02 | 1.06E-06 | 1.05E-06 |
| Am-243 | 3.62E+00 | 4.40E+02 | 8.88E-05 | 8.88E-05 |
| Ar-39 | | | 5.40E-07 | 5.31E-07 |
| At-217 | | | 1.11E-06 | 1.01E-06 |
| At-218 | | | 3.65E-06 | 3.65E-06 |
| Ba-133 | 3.40E-03 | 7.81E-03 | 1.24E-03 | 1.15E-03 |
| Ba-137m | | | 2.25E-03 | 2.00E-03 |
| Bi-210 | 6.39E-03 | 1.96E-01 | 2.25E-06 | 2.17E-06 |
| Bi-211 | | | 1.60E-04 | 1.49E-04 |
| Bi-212 | 1.06E-03 | 2.16E-02 | 7.32E-04 | 6.26E-04 |
| Bi-213 | 7.22E-04 | 1.71E-02 | 4.79E-04 | 4.38E-04 |
| Bi-214 | 2.83E-04 | 6.59E-03 | 6.13E-03 | 5.09E-03 |
| Bk-249 | 1.20E-02 | 1.39E+00 | 2.91E-09 | 2.90E-09 |
| C-14 | 2.09E-03 | 2.09E-03 | 8.41E-09 | 8.41E-09 |
| Ca-41 | 1.27E-03 | 1.35E-03 | 0.00E+00 | 0.00E+00 |
| Cd-113m | 1.61E-01 | 1.53E+00 | 4.05E-07 | 3.99E-07 |
| Cf-249 | 4.74E+00 | 5.77E+02 | 1.16E-03 | 1.07E-03 |
| Cf-250 | 2.13E+00 | 2.62E+02 | 7.40E-08 | 7.40E-08 |
| Cf-251 | 4.85E+00 | 5.88E+02 | 3.29E-04 | 3.22E-04 |
| Cf-252 | 1.08E+00 | 1.57E+02 | 1.10E-07 | 1.10E-07 |
| Cl-36 | 3.03E-03 | 2.19E-02 | 1.50E-06 | 1.42E-06 |
| Cm-242 | 1.15E-01 | 1.73E+01 | 1.07E-07 | 1.06E-07 |
| Cm-243 | 2.51E+00 | 3.07E+02 | 3.64E-04 | 3.53E-04 |
| Cm-244 | 2.02E+00 | 2.48E+02 | 7.87E-08 | 7.87E-08 |
| Cm-245 | 3.74E+00 | 4.55E+02 | 2.13E-04 | 2.10E-04 |
| Cm-246 | 3.70E+00 | 4.51E+02 | 7.26E-08 | 7.26E-08 |
| Cm-247 | 3.42E+00 | 4.14E+02 | 1.11E-03 | 1.03E-03 |
| Cm-248 | 1.36E+01 | 1.65E+03 | 5.49E-08 | 5.49E-08 |
| Co-60 | 2.69E-02 | 2.19E-01 | 1.01E-02 | 8.47E-03 |
| Cs-134 | 7.32E-02 | 4.64E-02 | 5.92E-03 | 5.22E-03 |
| Cs-135 | 7.07E-03 | 4.55E-03 | 2.39E-08 | 2.40E-08 |
| Cs-137 | 5.00E-02 | 3.19E-02 | 4.70E-07 | 4.60E-07 |

| Table B-2. Intruder Input – Internal and External Dose Conversion Factors (Lee 2004) | | | | |
|---|---|---|---|---|
| | Internal DCFs (rem/μCi) | | External DCFs (rem/yr per μCi/m$^3$) | |
| **Nuclide** | **Ingestion** | **Inhalation** | **Infinite Depth** | **15 cm** |
| Eu-152 | 6.48E-03 | 2.21E-01 | 4.38E-03 | 3.76E-03 |
| Eu-154 | 9.55E-03 | 2.86E-01 | 4.80E-03 | 4.11E-03 |
| Eu-155 | 1.53E-03 | 4.15E-02 | 1.14E-04 | 1.14E-04 |
| Fr-221 | | | 9.60E-05 | 9.23E-05 |
| Fr-223 | 8.62E-03 | 6.22E-03 | 1.24E-04 | 1.18E-04 |
| Gd-152 | 1.61E-01 | 2.43E+02 | 0.00E+00 | 0.00E+00 |
| H-3 | 6.40E-05 | 6.40E-05 | 0.00E+00 | 0.00E+00 |
| I-129 | 2.76E-01 | 1.74E-01 | 8.10E-06 | 8.10E-06 |
| K-40 | 1.86E-02 | 1.24E-02 | 6.51E-04 | 5.34E-04 |
| Kr-85 | | | 8.94E-06 | 8.14E-06 |
| Mo-93 | 1.35E-03 | 2.84E-02 | 3.69E-07 | 3.69E-07 |
| Nb-93m | 5.22E-04 | 2.92E-02 | 6.50E-08 | 6.50E-08 |
| Nb-94 | 7.14E-03 | 4.14E-01 | 6.05E-03 | 5.29E-03 |
| Ni-59 | 2.10E-04 | 1.32E-03 | 0.00E+00 | 0.00E+00 |
| Ni-63 | 5.77E-04 | 3.10E-03 | 0.00E+00 | 0.00E+00 |
| Np-237 | 4.44E+00 | 5.40E+02 | 4.87E-05 | 4.86E-05 |
| Np-238 | 4.00E-03 | 3.71E-02 | 2.15E-03 | 1.84E-03 |
| Np-239 | 3.26E-03 | 2.51E-03 | 4.71E-04 | 4.56E-04 |
| Np-240 | 2.37E-04 | 8.14E-05 | 4.83E-03 | 4.26E-03 |
| Np-240m | | | 1.26E-03 | 1.11E-03 |
| Pa-231 | 1.06E+01 | 1.28E+03 | 1.19E-04 | 1.12E-04 |
| Pa-233 | 3.63E-03 | 9.55E-03 | 6.38E-04 | 6.03E-04 |
| Pa-234 | 2.16E-03 | 8.14E-04 | 7.22E-03 | 6.28E-03 |
| Pa-234m | | | 5.61E-05 | 4.90E-05 |
| Pb-209 | 2.13E-04 | 9.49E-05 | 4.83E-07 | 4.76E-07 |
| Pb-210 | 5.37E+00 | 1.36E+01 | 1.53E-06 | 1.53E-06 |
| Pb-211 | 5.26E-04 | 8.71E-03 | 1.91E-04 | 1.70E-04 |
| Pb-212 | 4.55E-02 | 1.69E-01 | 4.40E-04 | 4.23E-04 |
| Pb-214 | 6.25E-04 | 7.81E-03 | 8.39E-04 | 7.83E-04 |
| Pd-107 | 1.49E-04 | 1.28E-02 | 0.00E+00 | 0.00E+00 |
| Po-210 | 1.90E+00 | 9.40E+00 | 3.27E-08 | 2.86E-08 |
| Po-211 | | | 2.98E-05 | 2.62E-05 |
| Po-212 | | | 0.00E+00 | 0.00E+00 |
| Po-213 | | | 0.00E+00 | 0.00E+00 |
| Po-214 | | | 3.21E-07 | 2.80E-07 |
| Po-215 | | | 6.35E-07 | 5.82E-07 |
| Po-216 | | | 6.52E-08 | 5.69E-08 |
| Po-218 | | | 3.53E-08 | 3.07E-08 |
| Pu-238 | 3.20E+00 | 3.92E+02 | 9.46E-08 | 9.43E-08 |

| Table B-2. Intruder Input – Internal and External Dose Conversion Factors (Lee 2004) | | | | |
|---|---|---|---|---|
| | **Internal DCFs (rem/μCi)** | | **External DCFs (rem/yr per μCi/m$^3$)** | |
| **Nuclide** | **Ingestion** | **Inhalation** | **Infinite Depth** | **15 cm** |
| Pu-239 | 3.54E+00 | 4.29E+02 | 1.85E-07 | 1.78E-07 |
| Pu-240 | 3.54E+00 | 4.29E+02 | 9.17E-08 | 9.16E-08 |
| Pu-241 | 6.85E-02 | 8.25E+00 | 3.69E-09 | 3.68E-09 |
| Pu-242 | 3.36E+00 | 4.11E+02 | 8.00E-08 | 8.00E-08 |
| Pu-243 | 3.34E-04 | 1.64E-04 | 4.98E-05 | 4.90E-05 |
| Pu-244 | 3.32E+00 | 4.03E+02 | 4.72E-08 | 4.72E-08 |
| Ra-223 | 6.59E-01 | 7.84E+00 | 3.77E-04 | 3.62E-04 |
| Ra-224 | 3.66E-01 | 3.16E+00 | 3.20E-05 | 3.06E-05 |
| Ra-225 | 3.85E-01 | 7.77E+00 | 6.89E-06 | 6.89E-06 |
| Ra-226 | 1.32E+00 | 8.58E+00 | 1.99E-05 | 1.93E-05 |
| Ra-228 | 1.44E+00 | 4.77E+00 | 0.00E+00 | 0.00E+00 |
| Rb-87 | 4.92E-03 | 3.23E-03 | 8.81E-08 | 8.78E-08 |
| Re-188 | 3.07E-03 | 2.01E-03 | 2.01E-04 | 1.83E-04 |
| Rn-219 | | | 1.93E-04 | 1.80E-04 |
| Rn-220 | | | 1.44E-06 | 1.28E-06 |
| Rn-222 | | | 1.47E-06 | 1.33E-06 |
| S-35 | 7.33E-04 | 2.48E-03 | 9.31E-09 | 9.31E-09 |
| Sb-126 | 1.07E-02 | 1.17E-02 | 1.07E-02 | 9.50E-03 |
| Sb-126m | 9.36E-05 | 3.39E-05 | 5.82E-03 | 5.19E-03 |
| Sc-46 | 6.40E-03 | 2.96E-02 | 7.93E-03 | 6.77E-03 |
| Se-79 | 8.70E-03 | 9.84E-03 | 1.16E-08 | 1.16E-08 |
| Sm-151 | 3.89E-04 | 3.00E-02 | 6.15E-10 | 6.15E-10 |
| Sn-121 | 9.02E-04 | 5.11E-04 | 1.23E-07 | 1.21E-07 |
| Sn-121m | 1.55E-03 | 1.15E-02 | 1.23E-06 | 1.23E-06 |
| Sn-126 | 1.95E-02 | 9.95E-02 | 9.22E-05 | 9.22E-05 |
| Sr-90 | 1.42E-01 | 1.30E+00 | 4.40E-07 | 4.34E-07 |
| Tc-99 | 1.46E-03 | 8.33E-03 | 7.85E-08 | 7.82E-08 |
| Th-227 | 3.81E-02 | 1.62E+01 | 3.26E-04 | 3.10E-04 |
| Th-228 | 3.96E-01 | 3.42E+02 | 4.96E-06 | 4.87E-06 |
| Th-229 | 3.53E+00 | 2.15E+03 | 2.01E-04 | 1.99E-04 |
| Th-230 | 5.48E-01 | 3.26E+02 | 7.56E-07 | 7.46E-07 |
| Th-231 | 1.35E-03 | 8.77E-04 | 2.28E-05 | 2.27E-05 |
| Th-232 | 2.73E+00 | 1.64E+03 | 3.26E-07 | 3.25E-07 |
| Th-234 | 1.37E-02 | 3.50E-02 | 1.51E-05 | 1.51E-05 |
| Tl-207 | | | 1.24E-05 | 1.11E-05 |
| Tl-208 | | | 1.44E-02 | 1.13E-02 |
| Tl-209 | | | 8.08E-03 | 6.76E-03 |
| U-232 | 1.31E+00 | 6.59E+02 | 5.64E-07 | 5.57E-07 |
| U-233 | 2.89E-01 | 1.35E+02 | 8.74E-07 | 8.46E-07 |

| Table B-2. Intruder Input – Internal and External Dose Conversion Factors (Lee 2004) | | | | |
|---|---|---|---|---|
| | **Internal DCFs (rem/μCi)** | | **External DCFs (rem/yr per μCi/m$^3$)** | |
| **Nuclide** | **Ingestion** | **Inhalation** | **Infinite Depth** | **15 cm** |
| U-234 | 2.83E-01 | 1.32E+02 | 2.51E-07 | 2.50E-07 |
| U-235 | 2.66E-01 | 1.23E+02 | 4.51E-04 | 4.38E-04 |
| U-236 | 2.69E-01 | 1.25E+02 | 1.34E-07 | 1.33E-07 |
| U-238 | 2.55E-01 | 1.18E+02 | 6.45E-08 | 6.45E-08 |
| U-240 | 4.45E-03 | 2.27E-03 | 8.90E-07 | 8.90E-07 |
| W-181 | 3.44E-04 | 1.51E-04 | 4.78E-05 | 4.78E-05 |
| W-185 | 1.99E-03 | 7.50E-04 | 2.71E-07 | 2.69E-07 |
| W-188 | 9.40E-03 | 4.09E-03 | 6.05E-06 | 5.75E-06 |
| Y-90 | 1.08E-02 | 8.44E-03 | 1.50E-05 | 1.40E-05 |
| Zr-93 | 1.66E-03 | 3.21E-01 | 0.00E+00 | 0.00E+00 |

**Table B-3. Intruder Analysis Computational Code Input – External Pathway Shielding Dose Coefficients ( Rem/yr per μCI/M3) (Lee 2004)**

| Radionuclide | 0 cm | 5 cm | 100 cm | Radionuclide | 0 cm | 5 cm | 100 cm |
|---|---|---|---|---|---|---|---|
| AC-225 | 3.51E-05 | 1.01E-05 | 1.05E-11 | AU-199 | 2.65E-04 | 8.53E-05 | 3.15E-12 |
| AC-227 | 3.66E-07 | 8.25E-08 | 1.52E-16 | BA-131 | 2.04E-03 | 8.96E-04 | 2.18E-08 |
| AC-228 | 4.66E-03 | 2.41E-03 | 7.19E-07 | BA-133 | 1.58E-03 | 6.57E-04 | 2.23E-09 |
| AG-106M | 1.42E-02 | 7.26E-03 | 2.04E-06 | BA-133M | 4.94E-04 | 8.31E-05 | 5.98E-11 |
| AG-108 | 8.77E-05 | 4.19E-05 | 1.40E-09 | BA-135M | 1.97E-04 | 6.92E-05 | 3.78E-11 |
| AG-108M | 8.08E-03 | 3.88E-03 | 1.57E-07 | BA-137M | 3.04E-03 | 1.49E-03 | 6.64E-08 |
| AG-109M | 1.14E-05 | 1.25E-06 | 3.05E-17 | BA-139 | 1.22E-04 | 4.65E-05 | 9.74E-09 |
| AG-110 | 1.55E-04 | 7.63E-05 | 3.89E-09 | BA-140 | 8.99E-04 | 4.12E-04 | 6.81E-09 |
| AG-110M | 1.41E-02 | 7.29E-03 | 1.80E-06 | BA-141 | 4.25E-03 | 2.07E-03 | 7.64E-07 |
| AG-111 | 1.21E-04 | 5.16E-05 | 1.92E-10 | BA-142 | 4.49E-03 | 2.31E-03 | 6.00E-07 |
| AL-26 | 1.36E-02 | 7.46E-03 | 7.87E-06 | BE-7 | 2.44E-04 | 1.12E-04 | 1.33E-09 |
| AL-28 | 9.18E-03 | 5.34E-03 | 7.22E-06 | BI-206 | 1.64E-02 | 8.44E-03 | 3.28E-06 |
| AM-241 | 2.70E-05 | 2.06E-06 | 7.90E-22 | BI-207 | 7.65E-03 | 3.92E-03 | 1.09E-06 |
| AM-242 | 3.31E-05 | 7.91E-06 | 7.27E-15 | BI-208 | 1.33E-02 | 8.16E-03 | 3.03E-05 |
| AM-242M | 2.83E-06 | 1.18E-07 | 9.06E-17 | BI-211 | 2.17E-04 | 9.34E-05 | 3.38E-10 |
| AM-243 | 7.90E-05 | 1.22E-05 | 3.76E-15 | BI-212 | 9.41E-04 | 4.94E-04 | 2.08E-07 |
| AM-244 | 4.02E-03 | 2.01E-03 | 1.88E-07 | BI-213 | 6.65E-04 | 3.01E-04 | 7.52E-09 |
| AM-245 | 1.02E-04 | 3.47E-05 | 9.01E-12 | BI-214 | 7.74E-03 | 4.25E-03 | 3.83E-06 |
| AM-246 | 5.02E-03 | 2.64E-03 | 6.38E-07 | BK-250 | 4.57E-03 | 2.41E-03 | 5.17E-07 |
| AR-41 | 6.63E-03 | 3.66E-03 | 1.86E-06 | BR-77 | 1.48E-03 | 6.67E-04 | 2.04E-08 |
| AS-72 | 9.01E-03 | 4.46E-03 | 1.23E-06 | BR-80 | 3.77E-04 | 1.81E-04 | 6.76E-09 |
| AS-73 | 5.82E-06 | 3.00E-07 | | BR-80M | 1.50E-05 | 4.27E-08 | |
| AS-74 | 3.84E-03 | 1.83E-03 | 5.15E-08 | BR-82 | 1.36E-02 | 6.96E-03 | 1.62E-06 |
| AS-76 | 2.18E-03 | 1.09E-03 | 3.10E-07 | BR-83 | 3.72E-05 | 1.75E-05 | 3.28E-10 |
| AS-77 | 3.62E-05 | 1.48E-05 | 1.31E-10 | BR-84 | 9.10E-03 | 5.30E-03 | 1.71E-05 |
| AT-211 | 6.65E-05 | 1.50E-05 | 2.24E-10 | BR-85 | 3.42E-04 | 1.79E-04 | 5.50E-08 |
| AT-217 | 1.20E-06 | 5.76E-07 | 1.70E-11 | C-11 | 5.06E-03 | 2.36E-03 | 3.78E-08 |
| AU-194 | 5.19E-03 | 2.72E-03 | 2.15E-06 | CA-45 | 1.91E-13 | | |
| AU-195 | 4.44E+01 | 1.83E-05 | 1.07E-14 | CA-47 | 5.48E-03 | 3.00E-03 | 1.44E-06 |
| AU-195M | 7.58E-04 | 2.86E-04 | 1.21E-10 | CA-49 | 1.58E-02 | 1.00E-02 | 5.60E-05 |
| AU-196 | 2.05E-03 | 8.73E-04 | 4.67E-09 | CD-109 | 1.03E-05 | 1.88E-13 | |
| AU-198 | 1.94E-03 | 8.65E-04 | 8.07E-09 | CD-111M | 1.05E-03 | 3.80E-04 | 7.95E-11 |
| | | | | | | | |

**Table B-3. Intruder Analysis Computational Code Input – External Pathway Shielding Dose Coefficients ( Rem/yr per μCI/M$^{3)}$ (Lee 2004)**

| Radionuclide | 0 cm | 5 cm | 100 cm | Radionuclide | 0 cm | 5 cm | 100 cm |
|---|---|---|---|---|---|---|---|
| CD-115 | 1.00E-03 | 4.68E-04 | 7.95E-09 | CR-49 | 4.92E-03 | 2.25E-03 | 3.86E-06 |
| CD-115M | 1.13E-04 | 5.99E-05 | 1.69E-08 | CR-51 | 1.45E-04 | 6.13E-05 | 1.43E-10 |
| CD-117 | 5.50E-03 | 2.92E-03 | 1.58E-06 | CS-126 | 5.57E-03 | 2.62E-03 | 2.52E-07 |
| CD-117M | 1.04E-02 | 5.86E-03 | 7.19E-06 | CS-129 | 1.22E-03 | 5.30E-04 | 4.54E-09 |
| CE-139 | 4.53E-04 | 1.44E-04 | 4.67E-12 | CS-131 | 1.68E-05 | 3.83E-09 | |
| CE-141 | 2.22E-04 | 6.98E-05 | 1.21E-12 | CS-132 | 3.55E-03 | 1.74E-03 | 9.69E-08 |
| CE-143 | 1.17E-03 | 5.11E-04 | 1.73E-08 | CS-134 | 7.95E-03 | 3.95E-03 | 3.33E-07 |
| CE-144 | 4.72E-05 | 1.35E-05 | 1.19E-13 | CS-134M | 5.43E-05 | 1.41E-05 | 8.15E-14 |
| CF-248 | 6.45E-07 | 6.99E-11 | | CS-136 | 1.09E-02 | 5.61E-03 | 1.09E-06 |
| CF-249 | 1.53E-03 | 6.65E-04 | 3.05E-09 | CS-138 | 1.21E-02 | 6.74E-03 | 8.10E-06 |
| CF-250 | 6.66E-07 | 6.13E-09 | 5.00E-21 | CS-139 | 1.57E-03 | 8.97E-04 | 1.29E-06 |
| CF-251 | 3.43E-04 | 1.08E-04 | 1.51E-11 | CU-61 | 4.08E-03 | 1.92E-03 | 9.64E-08 |
| CF-252 | 6.12E-07 | 3.08E-09 | 5.40E-23 | CU-62 | 5.00E-03 | 2.33E-03 | 5.17E-08 |
| CF-253 | 1.09E-08 | 8.03E-12 | | CU-64 | 9.39E-04 | 4.40E-04 | 1.77E-08 |
| CF-254 | 1.82E-11 | 2.23E-13 | | CU-67 | 3.60E-04 | 1.18E-04 | 2.09E-11 |
| CL-38 | 7.64E-03 | 4.52E-03 | 8.55E-06 | DY-157 | 1.48E-03 | 6.15E-04 | 1.77E-09 |
| CM-242 | 8.66E-07 | 2.13E-09 | | DY-165 | 9.90E-05 | 4.35E-05 | 1.46E-09 |
| CM-243 | 4.15E-04 | 1.42E-04 | 5.80E-11 | DY-166 | 6.02E-05 | 1.33E-05 | 5.40E-11 |
| CM-244 | 7.68E-07 | 1.24E-09 | | ER-169 | 3.93E-09 | 1.06E-09 | 1.11E-18 |
| CM-245 | 1.79E-04 | 4.93E-05 | 5.55E-13 | ER-l71 | 1.53E-03 | 6.23E-04 | 5.53E-09 |
| CM-246 | 6.80E-07 | 1.76E-10 | | ES-253 | 1.40E-06 | 3.50E-07 | 2.14E-14 |
| CM-247 | 1.50E-03 | 6.56E-04 | 3.61E-09 | ES-254 | 1.56E-05 | 2.33E-06 | 2.10E-12 |
| CM-248 | 5.45E-07 | 1.07E-09 | | ES-254M | 2.84E-03 | 1.39E-03 | 6.56E-08 |
| CM-249 | 9.37E-05 | 4.46E-05 | 1.37E-09 | ES-255 | 4.84E-08 | 6.26E-12 | |
| CO-56 | 1.85E-02 | 1.03E-02 | 1.83E-05 | EU-152 | 5.66E-03 | 2.93E-03 | 9.86E-07 |
| CO-57 | 3.37E-04 | 1.00E-04 | 1.56E-10 | EU-152M | 1.56E-03 | 7.98E-04 | 1.45E-07 |
| CO-58 | 5.00E-03 | 2.51E-03 | 2.44E-07 | EU-154 | 6.27E-03 | 3.28E-03 | 1.04E-06 |
| CO-58M | 7.45E-09 | 6.56E-16 | | EU-155 | 1.09E-04 | 2.32E-05 | 7.06E-15 |
| CO-60 | 1.30E-02 | 7.13E-03 | 3.33E-06 | EU-156 | 6.87E-03 | 3.81E-03 | 3.96E-06 |
| CO-60M | 1.82E-05 | 9.51E-06 | 5.17E-09 | F-18 | 4.91E-03 | 2.29E-03 | 3.68E-08 |
| CO-61 | 2.44E-04 | 9.74E-05 | 1.23E-08 | FE-52 | 3.37E-03 | 1.50E-03 | 2.13E-08 |

**Table B-3. Intruder Analysis Computational Code Input – External Pathway Shielding Dose Coefficients ( Rem/yr per μCI/M$^{3}$) (Lee 2004)**

| Radionuclide | 0 cm | 5 cm | 100 cm | Radionuclide | 0 cm | 5 cm | 100 cm |
|---|---|---|---|---|---|---|---|
| FE-59 | 6.13E-03 | 3.33E-03 | 1.31E-06 | I-133 | 3.05E-03 | 1.47E-03 | 1.17E-07 |
| FM-254 | 7.56E-07 | 1.34E-08 | 1.50E-19 | I-134 | 1.35E-02 | 7.03E-03 | 2.10E-06 |
| FM-255 | 8.03E-06 | 5.23E-07 | 3.20E-15 | I-135 | 8.10E-03 | 4.46E-03 | 3.15E-06 |
| FR-221 | 1.07E-04 | 3.71E-05 | 8.70E-12 | I-136 | 1.29E-02 | 7.51E-03 | 1.78E-05 |
| FR-223 | 1.32E-04 | 4.43E-05 | 8.28E-10 | IN-111 | 1.42E-03 | 5.02E-04 | 8.45E-11 |
| GA-66 | 1.25E-02 | 7.12E-03 | 2.45E-05 | IN-113M | 1.22E-03 | 5.35E-04 | 2.83E-09 |
| GA-67 | 5.19E-04 | 1.94E-04 | 8.53E-10 | IN-114 | 1.60E-04 | 7.75E-05 | 5.63E-09 |
| GA-68 | 4.72E-03 | 2.22E-03 | 7.49E-08 | IN-114M | 4.03E-04 | 1.78E-04 | 6.69E-09 |
| GA-72 | 1.39E-02 | 7.74E-03 | 1.15E-05 | IN-115M | 7.41E-04 | 3.16E-04 | 9.64E-10 |
| GD-153 | 1.65E-04 | 2.96E-05 | 7.52E-15 | IN-116M | 1.26E-02 | 6.94E-03 | 4.79E-06 |
| GD-159 | 1.60E-04 | 6.67E-05 | 2.72E-10 | IN-117 | 3.19E-03 | 1.45E-03 | 2.90E-08 |
| GD-162 | 2.03E-03 | 9.08E-04 | 6.56E-09 | IN-117M | 3.66E-04 | 1.46E-04 | 3.81E-10 |
| GE-68 | 1.62E-07 | | | IR-190 | 6.55E-03 | 3.01E-03 | 1.16E-07 |
| GE-71 | 1.64E-07 | | | IR-190M | 2.05E-11 | 1.74E-17 | |
| GE-77 | 5.05E-03 | 2.42E-03 | 6.86E-07 | IR-190M2 | 5.64E-05 | 6.39E-06 | 3.18E-16 |
| HF-181 | 2.47E-03 | 1.09E-03 | 1.17E-08 | IR-192 | 3.82E-03 | 1.68E-03 | 1.72E-08 |
| HG-197 | 9.35E-05 | 1.46E-05 | 1.48E-13 | IR-193M | 4.13E-07 | 5.08E-08 | 7.39E-21 |
| HG-197M | 2.37E-04 | 7.20E-05 | 1.91E-11 | IR-194 | 4.38E-04 | 2.06E-04 | 2.80E-08 |
| HG-203 | 9.35E-04 | 3.68E-04 | 2.93E-10 | IR-194M | 1.14E-02 | 5.33E-03 | 1.43E-07 |
| HO-166 | 1.09E-04 | 5.58E-05 | 3.89E-08 | K-40 | 8.05E-04 | 4.54E-04 | 3.41E-07 |
| HO-166M | 7.71E-03 | 3.72E-03 | 2.95E-07 | K-42 | 1.42E-03 | 8.07E-04 | 6.89E-07 |
| I-122 | 4.83E-03 | 2.28E-03 | 2.22E-07 | K-43 | 4.78E-03 | 2.23E-03 | 6.74E-08 |
| I-123 | 5.21E-04 | 1.76E-04 | 5.90E-10 | KR-79 | 1.20E-03 | 5.51E-04 | 2.57E-08 |
| I-124 | 5.33E-03 | 2.73E-03 | 1.49E-06 | KR-81 | 4.44E-05 | 1.66E-05 | 1.19E-11 |
| I-125 | 3.05E-05 | 4.64E-09 | | KR-83M | 8.95E-07 | 4.72E-12 | |
| I-126 | 2.28E-03 | 1.09E-03 | 5.27E-08 | KR-85 | 1.11E-05 | 5.18E-06 | 8.53E-11 |
| I-128 | 3.65E-04 | 1.67E-04 | 3.41E-09 | KR-85M | 5.51E-04 | 1.97E-04 | 1.46E-10 |
| I-129 | 1.83E-05 | 1.84E-08 | | KR-87 | 3.98E-03 | 2.22E-03 | 4.77E-06 |
| I-130 | 1.08E-02 | 5.32E-03 | 3.78E-07 | KR-88 | 9.90E-03 | 5.80E-03 | 1.27E-05 |
| I-131 | 1.81E-03 | 8.02E-04 | 9.28E-09 | KR-89 | 9.24E-03 | 5.20E-03 | 1.20E-05 |
| I-132 | 1.17E-02 | 5.95E-03 | 1.19E-06 | KR-90 | 6.37E-03 | 3.41E-03 | 2.75E-06 |

**Table B-3. Intruder Analysis Computational Code Input – External Pathway Shielding Dose Coefficients ( Rem/yr per μCI/M$^{3}$) (Lee 2004)**

| Radionuclide | 0 cm | 5 cm | 100 cm | Radionuclide | 0 cm | 5 cm | 100 cm |
|---|---|---|---|---|---|---|---|
| LA-140 | 1.19E-02 | 6.51E-03 | 5.68E-06 | NB-97 | 3.39E-03 | 1.66E-03 | 8.53E-08 |
| LA-141 | 2.20E-04 | 1.23E-04 | 8.45E-08 | NB-97M | 3.75E-03 | 1.87E-03 | 1.31E-07 |
| LA-142 | 1.39E-02 | 8.09E-03 | 2.11E-05 | ND-147 | 5.39E-04 | 2.32E-04 | 4.16E-09 |
| LU-l77 | 1.08E-04 | 3.54E-05 | 5.45E-12 | ND-149 | 1.59E-03 | 6.71E-04 | 1.64E-08 |
| LU-l77M | 3.78E-03 | 1.48E-03 | 4.39E-09 | NI-56 | 8.42E-03 | 4.15E-03 | 8.45E-07 |
| MG-27 | 4.61E-03 | 2.39E-03 | 3.53E-07 | NI-57 | 9.77E-03 | 5.31E-03 | 3.91E-06 |
| MG-28 | 6.94E-03 | 3.71E-03 | 1.65E-06 | NI-65 | 2.83E-03 | 1.56E-03 | 9.86E-07 |
| MN-52 | 1.78E-02 | 9.43E-03 | 3.66E-06 | NP-235 | 6.23E-06 | 9.34E-07 | 4.21E-16 |
| MN-52M | 1.22E-02 | 6.40E-03 | 2.95E-06 | NP-236 | 3.32E-04 | 9.24E-05 | 1.43E-12 |
| MN-54 | 4.32E-03 | 2.21E-03 | 2.40E-07 | NP-236M | 1.44E-04 | 4.91E-05 | 1.21E-09 |
| MN-56 | 8.72E-03 | 4.83E-03 | 5.12E-06 | NP-237 | 5.09E-05 | 1.09E-05 | 1.34E-13 |
| MN-57 | 3.29E-04 | 1.57E-04 | 4.39E-08 | NP-238 | 2.84E-03 | 1.50E-03 | 3.13E-07 |
| MO-101 | 7.57E-03 | 4.05E-03 | 2.78E-06 | NP-239 | 5.36E-04 | 1.83E-04 | 1.20E-10 |
| MO-91 | 4.85E-03 | 2.27E-03 | 1.52E-07 | NP-240 | 5.67E-03 | 2.81E-03 | 3.61E-07 |
| MO-93 | 6.28E-06 | 2.78E-20 | | NP-240M | 1.66E-03 | 8.20E-04 | 1.43E-07 |
| MO-99 | 7.60E-04 | 3.70E-04 | 2.49E-08 | O-15 | 5.07E-03 | 2.36E-03 | 3.78E-08 |
| N-13 | 5.06E-03 | 2.36E-03 | 3.78E-08 | OS-185 | 3.41E-03 | 1.66E-03 | 9.28E-08 |
| N-16 | 2.16E-02 | 1.45E-02 | 2.28E-04 | OS-190M | 7.58E-03 | 3.48E-03 | 7.01E-08 |
| NA-22 | 1.11E-02 | 5.74E-03 | 1.79E-06 | OS-191 | 1.43E-04 | 3.50E-05 | 1.98E-13 |
| NA-24 | 2.10E-02 | 1.26E-02 | 3.83E-05 | OS-191M | 5.47E-06 | 6.20E-07 | 2.30E-20 |
| NB-90 | 2.10E-02 | 1.19E-02 | 2.07E-05 | OS-193 | 2.69E-04 | 1.13E-04 | 1.12E-09 |
| NB-91 | 1.41E-05 | 3.87E-06 | 6.23E-11 | PA-230 | 3.17E-03 | 1.58E-03 | 2.21E-07 |
| NB-91M | 2.24E-04 | 1.20E-04 | 4.77E-08 | PA-231 | 1.26E-04 | 4.90E-05 | 7.97E-11 |
| NB-92 | 7.67E-03 | 3.87E-03 | 4.47E-07 | PA-233 | 8.69E-04 | 3.50E-04 | 7.90E-10 |
| NB-92M | 4.97E-03 | 2.59E-03 | 4.89E-07 | PA-234 | 9.68E-03 | 4.91E-03 | 1.18E-06 |
| NB-93M | 1.12E-06 | 4.97E-21 | | PA-234M | 5.75E-05 | 2.98E-05 | 5.20E-09 |
| NB-94 | 8.10E-03 | 4.11E-03 | 3.96E-07 | PB-203 | 3.62E-03 | 2.94E-03 | 1.02E-09 |
| NB-94M | 2.56E-05 | 1.10E-05 | 1.40E-09 | PB-204M | 1.06E-02 | 5.36E-03 | 6.81E-07 |
| NB-95 | 3.94E-03 | 1.98E-03 | 1.56E-07 | PB-205 | 6.74E-07 | | |
| NB-95M | 2.28E-04 | 8.16E-05 | 1.57E-11 | PB-210 | 2.51E-06 | 3.87E-08 | |
| NB-96 | 1.26E-02 | 6.44E-03 | 1.10E-06 | PB-211 | 2.55E-04 | 1.24E-04 | 8.30E-09 |

| **Table B-3. Intruder Analysis Computational Code Input – External Pathway Shielding Dose Coefficients ( Rem/yr per μCI/M$^{3)}$ (Lee 2004)** | | | | | | | |
|---|---|---|---|---|---|---|---|
| Radionuclide | 0 cm | 5 cm | 100 cm | Radionuclide | 0 cm | 5 cm | 100 cm |
| PB-212 | 4.99E-04 | 1.76E-04 | 5.58E-11 | PU-238 | 7.77E-07 | 1.73E-09 | |
| PB-214 | 1.09E-03 | 4.59E-04 | 4.84E-09 | PU-239 | 4.22E-07 | 3.74E-08 | 5.25E-17 |
| PD-103 | 1.00E-05 | 2.29E-07 | 9.11E-13 | PU-240 | 7.43E-07 | 1.71E-09 | |
| PD-109 | 3.41E-06 | 1.59E-06 | 2.51E-11 | PU-242 | 6.18E-07 | 1.85E-09 | |
| PM-143 | 1.48E-03 | 7.31E-04 | 5.10E-08 | PU-243 | 4.83E-05 | 1.24E-05 | 2.51E-11 |
| PM-144 | 7.77E-03 | 3.76E-03 | 1.56E-07 | PU-244 | 5.26E-07 | 1.60E-10 | |
| PM-145 | 2.60E-05 | 5.41E-07 | 5.12E-20 | PU-245 | 1.99E-03 | 9.42E-04 | 7.24E-08 |
| PM-146 | 3.71E-03 | 1.78E-03 | 8.53E-08 | PU-246 | 3.20E-04 | 1.08E-04 | 1.05E-11 |
| PM-147 | 9.45E-09 | 2.74E-09 | 9.28E-18 | RA-222 | 4.21E-05 | 1.79E-05 | 4.64E-11 |
| PM-148 | 2.94E-03 | 1.57E-03 | 7.75E-07 | RA-223 | 4.47E-04 | 1.63E-04 | 2.95E-10 |
| PM-148M | 1.00E-02 | 4.88E-03 | 3.15E-07 | RA-224 | 3.76E-05 | 1.37E-05 | 4.67E-12 |
| PM-149 | 5.20E-05 | 2.22E-05 | 4.29E-10 | RA-225 | 1.09E-05 | 6.79E-08 | |
| PM-151 | 1.47E-03 | 6.44E-04 | 1.81E-08 | RA-226 | 2.16E-05 | 7.21E-06 | 3.89E-13 |
| PO-209 | 1.59E-05 | 7.63E-06 | 8.65E-10 | RB-81 | 2.81E-03 | 1.25E-03 | 2.35E-08 |
| PO-210 | 4.40E-08 | 2.23E-08 | 2.10E-12 | RB-82 | 5.46E-03 | 2.57E-03 | 8.93E-08 |
| PO-211 | 3.97E-05 | 1.99E-05 | 1.97E-09 | RB-83 | 2.49E-03 | 1.17E-03 | 2.32E-08 |
| PO-213 | 1.57E-07 | 7.93E-08 | 6.66E-12 | RB-84 | 4.58E-03 | 2.30E-03 | 3.10E-07 |
| PO-214 | 4.30E-07 | 2.18E-07 | 2.00E-11 | RB-86 | 7.09E+01 | 7.09E+01 | 7.09E+01 |
| PO-215 | 7.23E-07 | 3.25E-07 | 2.62E-12 | RB-88 | 3.27E-03 | 1.88E-03 | 3.23E-06 |
| PO-216 | 7.49E-08 | 3.80E-08 | 3.61E-12 | RB-89 | 1.06E-02 | 5.98E-03 | 8.30E-06 |
| PR-142 | 3.01E-04 | 1.72E-04 | 1.62E-07 | RB-90 | 1.07E-02 | 6.66E-03 | 4.64E-05 |
| PR-143 | 4.58E-11 | 2.29E-11 | 1.59E-15 | RB-90M | 1.66E-02 | 9.67E-03 | 3.18E-05 |
| PR-144 | 1.63E-04 | 9.16E-05 | 1.36E-07 | RE-182 | 2.65E+01 | 4.01E-03 | 1.40E-06 |
| PR-144M | 8.73E-06 | 3.57E-08 | | RE-182M | 5.72E-03 | 3.04E-03 | 1.37E-06 |
| PT-191 | 1.04E-03 | 4.26E-04 | 5.53E-09 | RE-183 | 3.37E-04 | 9.53E-05 | 4.31E-11 |
| PT-193M | 1.38E-05 | 1.89E-06 | 1.47E-15 | RE-184 | 4.32E-03 | 2.18E-03 | 2.65E-07 |
| PT-195M | 1.05E-04 | 1.85E-05 | 3.86E-14 | RE-184M | 1.61E-03 | 7.43E-04 | 8.60E-08 |
| PT-197 | 4.87E-05 | 1.27E-05 | 1.03E-12 | RE-186 | 5.03E-05 | 1.46E-05 | 5.63E-11 |
| PT-197M | 2.58E-04 | 9.75E-05 | 2.88E-10 | RE-188 | 2.38E-04 | 1.06E-04 | 1.17E-08 |
| PU-236 | 9.03E-07 | 5.27E-09 | | RH-103M | 1.07E-06 | 1.45E-10 | |
| PU-237 | 1.07E-04 | 2.56E-05 | 1.66E-14 | RH-105 | 3.55E-04 | 1.50E-04 | 3.23E-10 |
| RH-105M | 8.18E-05 | 2.31E-05 | 1.58E-13 | SR-85 | 2.54E-03 | 1.18E-03 | 1.95E-08 |
| RH-106 | 1.04E-03 | 5.06E-04 | 4.72E-08 | SR-85M | 7.96E-04 | 2.85E-04 | 3.89E-11 |

**Table B-3. Intruder Analysis Computational Code Input – External Pathway Shielding Dose Coefficients ( Rem/yr per μCI/M$^{3)}$ (Lee 2004)**

| Radionuclide | 0 cm | 5 cm | 100 cm | Radionuclide | 0 cm | 5 cm | 100 cm |
|---|---|---|---|---|---|---|---|
| RN-218 | 3.82E-06 | 1.84E-06 | 5.98E-11 | SR-87M | 1.53E-03 | 6.73E-04 | 3.43E-09 |
| RN-219 | 2.51E-04 | 1.05E-04 | 3.53E-10 | SR-89 | 7.07E-07 | 3.67E-07 | 5.45E-11 |
| RN-220 | 2.62E-06 | 1.24E-06 | 2.70E-11 | SR-91 | 3.54E-03 | 1.83E-03 | 3.43E-07 |
| RN-222 | 1.93E-06 | 9.02E-07 | 1.46E-11 | SR-92 | 6.90E-03 | 3.83E-03 | 2.32E-06 |
| RU-103 | 2.39E-03 | 1.11E-03 | 1.74E-08 | SR-93 | 1.13E-02 | 6.02E-03 | 4.79E-06 |
| RU-105 | 3.93E-03 | 1.91E-03 | 1.12E-07 | TA-182 | 6.34E-03 | 3.36E-03 | 1.22E-06 |
| RU-97 | 8.71E-04 | 3.19E-04 | 8.35E-10 | TB-157 | 3.55E-06 | 5.92E-08 | |
| SB-117 | 5.89E-04 | 2.11E-04 | 5.88E-09 | TB-160 | 5.42E-03 | 2.80E-03 | 6.66E-07 |
| SB-122 | 2.23E-03 | 1.07E-03 | 4.77E-08 | TB-162 | 5.32E-03 | 2.60E-03 | 3.03E-07 |
| SB-124 | 9.58E-03 | 5.16E-03 | 3.99E-06 | TC-101 | 1.57E-03 | 6.71E-04 | 6.43E-09 |
| SB-125 | 2.06E-03 | 9.54E-04 | 2.17E-08 | TC-95 | 4.05E-03 | 2.04E-03 | 1.86E-07 |
| SB-126 | 1.40E-02 | 6.84E-03 | 4.26E-07 | TC-95M | 3.20E-03 | 1.53E-03 | 1.19E-07 |
| SB-126M | 7.93E-03 | 3.82E-03 | 1.85E-07 | TC-96 | 1.29E-02 | 6.58E-03 | 7.64E-07 |
| SB-127 | 3.31E-03 | 1.60E-03 | 8.40E-08 | TC-96M | 2.17E-04 | 1.11E-04 | 2.47E-08 |
| SB-129 | 7.34E-03 | 3.82E-03 | 1.26E-06 | TC-97 | 7.12E-06 | 8.35E-19 | |
| SC-44 | 1.08E-02 | 5.52E-03 | 1.23E-06 | TC-97M | 6.44E-06 | 1.53E-07 | 2.04E-17 |
| SC-46 | 1.04E-02 | 5.52E-03 | 1.30E-06 | TC-98 | 7.14E-03 | 3.53E-03 | 2.04E-07 |
| SC-46M | 2.73E-04 | 8.70E-05 | 1.32E-12 | TC-99 | 9.63E-10 | 2.06E-10 | 6.81E-21 |
| SC-47 | 3.48E-04 | 1.15E-04 | 3.20E-12 | TC-99M | 3.81E-04 | 1.20E-04 | 1.65E-12 |
| SC-48 | 1.73E-02 | 9.31E-03 | 3.23E-06 | TE-121 | 2.81E-03 | 1.33E-03 | 3.18E-08 |
| SC-49 | 5.37E-06 | 3.12E-06 | 3.96E-09 | TE-121M | 7.78E-04 | 2.96E-04 | 2.36E-08 |
| SE-73 | 5.16E-03 | 2.33E-03 | 3.71E-08 | TE-123 | 9.52E-06 | 1.09E-10 | |
| SE-75 | 1.50E-03 | 5.73E-04 | 7.82E-10 | TE-123M | 4.39E-04 | 1.41E-04 | 3.91E-12 |
| SI-31 | 4.58E-06 | 2.52E-06 | 1.19E-09 | TE-125M | 2.62E-05 | 2.07E-07 | 1.81E-16 |
| SM-151 | 4.33E-09 | 4.04E-19 | | TE-127 | 2.29E-05 | 1.01E-05 | 6.28E-11 |
| SM-153 | 1.01E-04 | 2.08E-05 | 1.75E-11 | TE-127M | 8.64E-06 | 2.29E-07 | 7.49E-12 |
| SN-113 | 3.27E-05 | 7.49E-06 | 3.13E-12 | TE-129 | 2.62E-04 | 1.21E-04 | 6.96E-09 |
| SN-117M | 4.61E-04 | 1.48E-04 | 4.04E-12 | TE-129M | 1.67E-04 | 7.95E-05 | 4.52E-09 |
| SN-119M | 8.01E-06 | 1.92E-09 | 7.72E-24 | TE-131 | 1.91E-03 | 8.94E-04 | 1.22E-07 |
| SN-123 | 3.57E-05 | 1.91E-05 | 5.30E-09 | TE-131M | 7.18E-03 | 3.68E-03 | 1.08E-06 |
| SN-125 | 1.56E-03 | 8.39E-04 | 4.39E-07 | TE-132 | 7.74E-04 | 2.70E-04 | 3.41E-11 |
| SN-126 | 8.67E-05 | 1.59E-05 | 3.20E-16 | TE-133 | 4.61E-03 | 2.31E-03 | 9.28E-07 |
| SR-82 | 3.76E-06 | | | TE-133M | 1.12E-02 | 5.79E-03 | 2.04E-06 |
| TE-134 | 4.12E-03 | 1.92E-03 | 9.74E-08 | W-181 | 4.21E-05 | 3.75E-06 | 3.99E-15 |

| Table B-3. Intruder Analysis Computational Code Input – External Pathway Shielding Dose Coefficients ( Rem/yr per μCI/M3) (Lee 2004) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Radionuclide | 0 cm | 5 cm | 100 cm | Radionuclide | 0 cm | 5 cm | 100 cm |
| TH-226 | 2.24E-05 | 6.86E-06 | 8.80E-13 | W-185 | 7.40E-08 | 2.20E-08 | 1.08E-16 |
| TH-227 | 3.97E-04 | 1.50E-04 | 1.57E-10 | W-187 | 2.28E-03 | 1.08E-03 | 4.39E-08 |
| TH-228 | 5.20E-06 | 1.36E-06 | 6.53E-14 | W-188 | 7.10E-06 | 2.77E-06 | 2.46E-12 |
| TH-229 | 2.02E-04 | 5.38E-05 | 1.44E-12 | XE-122 | 2.52E-04 | 9.87E-05 | 3.68E-10 |
| TH-230 | 1.12E-06 | 1.68E-07 | 4.26E-15 | XE-123 | 2.90E-03 | 1.41E-03 | 5.78E-07 |
| TH-231 | 2.63E-05 | 4.14E-06 | 9.26E-15 | XE-125 | 9.55E-04 | 3.70E-04 | 1.96E-08 |
| TH-232 | 6.90E-07 | 5.38E-08 | 2.07E-16 | XE-127 | 9.71E-04 | 3.55E-04 | 6.43E-10 |
| TH-233 | 1.50E-04 | 6.70E-05 | 1.75E-09 | XE-129M | 6.33E-05 | 1.10E-05 | 5.85E-13 |
| TH-234 | 1.41E-05 | 2.72E-06 | 4.06E-16 | XE-131M | 2.27E-05 | 3.45E-06 | 1.07E-13 |
| TI-44 | 1.99E-04 | 3.09E-05 | 2.80E-15 | XE-133 | 5.96E-05 | 9.01E-06 | 5.65E-15 |
| TI-45 | 4.32E-03 | 2.02E-03 | 3.51E-08 | XE-133M | 1.03E-04 | 3.26E-05 | 4.97E-12 |
| TI-51 | 1.73E-03 | 7.69E-04 | 2.95E-08 | XE-135 | 9.92E-04 | 3.82E-04 | 1.68E-09 |
| TL-200 | 6.42E-03 | 3.26E-03 | 1.02E-06 | XE-135M | 2.13E-03 | 9.96E-04 | 1.82E-08 |
| TL-201 | 1.62E-04 | 3.62E-05 | 6.51E-13 | XE-137 | 9.25E-04 | 4.46E-04 | 1.69E-07 |
| TL-202 | 2.06E-03 | 9.03E-04 | 7.92E-09 | XE-138 | 5.62E-03 | 3.12E-03 | 4.36E-06 |
| TL-204 | 1.49E-06 | 2.24E-07 | 2.25E-19 | Y-86 | 1.83E-02 | 9.76E-03 | 6.71E-06 |
| TL-207 | 1.12E-05 | 5.82E-06 | 8.22E-10 | Y-87 | 2.26E-03 | 1.04E-03 | 1.32E-08 |
| TL-208 | 1.70E-02 | 9.98E-03 | 3.03E-05 | Y-88 | 1.38E-02 | 7.81E-03 | 8.68E-06 |
| TL-209 | 1.05E-02 | 5.67E-03 | 4.29E-06 | Y-90M | 2.84E-03 | 1.22E-03 | 1.22E-08 |
| TL-210 | 1.40E-02 | 7.52E-03 | 7.32E-06 | Y-91 | 1.87E-05 | 1.02E-05 | 4.09E-09 |
| TM-170 | 7.01E-06 | 1.06E-06 | 8.65E-18 | Y-91M | 2.66E-03 | 1.26E-03 | 2.90E-08 |
| TM-l71 | 6.68E-07 | 4.77E-08 | 1.25E-22 | Y-92 | 1.29E-03 | 6.83E-04 | 2.34E-07 |
| U-230 | 3.43E-06 | 8.20E-07 | 6.21E-14 | Y-93 | 4.40E-04 | 2.34E-04 | 2.26E-07 |
| U-231 | 1.49E-04 | 3.43E-05 | 3.28E-13 | YB-169 | 7.61E-04 | 2.32E-04 | 1.27E-10 |
| U-232 | 1.14E-06 | 1.10E-07 | 1.48E-15 | YB-175 | 1.70E-04 | 7.16E-05 | 3.10E-10 |
| U-233 | 7.65E-07 | 1.49E-07 | 2.51E-16 | ZN-62 | 2.20E-03 | 1.03E-03 | 2.33E-08 |
| U-234 | 8.02E-07 | 4.20E-08 | 1.33E-16 | ZN-65 | 3.00E-03 | 1.61E-03 | 4.84E-07 |
| U-235 | 4.84E-04 | 1.60E-04 | 6.51E-12 | ZN-69 | 2.88E-08 | 1.28E-08 | 8.20E-14 |
| U-236 | 6.76E-07 | 1.27E-08 | 2.28E-22 | ZN-69M | 2.02E-03 | 9.09E-04 | 7.34E-09 |
| U-237 | 3.60E-04 | 1.07E-04 | 3.30E-11 | ZR-86 | 1.10E-03 | 4.22E-04 | 3.03E-09 |
| U-238 | 5.92E-07 | 9.74E-09 | 6.11E-23 | ZR-88 | 1.83E-03 | 8.07E-04 | 4.29E-09 |
| U-239 | 1.06E-04 | 3.37E-05 | 1.44E-09 | ZR-89 | 5.92E-03 | 3.02E-03 | 4.34E-07 |
| U-240 | 3.59E-06 | 9.52E-09 | | ZR-95 | 3.78E-03 | 1.89E-03 | 1.32E-07 |
| V-48 | 1.49E-02 | 7.91E-03 | 2.95E-06 | ZR-97 | 9.16E-04 | 4.77E-04 | 2.15E-07 |
| V-52 | 7.45E-03 | 4.19E-03 | 2.95E-06 | | | | |

# Appendix C. Sensitivity Analysis

**Table C-1. C-14 predicted peak fractional flux to the water table and peak concentration at the 100 meter compliance well for Key Radionuclides scenario runs.**

| Scenario Run | Peak Fractional Flux (mole/yr/mole) | Peak Time (years) | Peak Concentration (pCi/L/Ci) | Peak Time (years) |
|---|---|---|---|---|
| 1 | 3.44E-24 | 1.00E+04 | 1.18E-19 | 1.00E+04 |
| 2 | 1.06E-25 | 1.00E+04 | 3.69E-21 | 1.00E+04 |
| 3 | 7.37E-23 | 1.00E+04 | 2.48E-18 | 1.00E+04 |
| 4 | 1.00E-25 | 1.00E+04 | 3.50E-21 | 1.00E+04 |
| 5 | 1.12E-20 | 1.00E+04 | 3.83E-16 | 1.00E+04 |
| 6 | 1.00E-25 | 1.00E+04 | 3.51E-21 | 1.00E+04 |
| 7 | 1.42E-20 | 1.00E+04 | 4.88E-16 | 1.00E+04 |
| 8 | 6.18E-25 | 1.00E+04 | 2.13E-20 | 1.00E+04 |
| 9 | 1.24E-22 | 1.00E+04 | 4.17E-18 | 1.00E+04 |
| 10 | 5.78E-25 | 1.00E+04 | 1.99E-20 | 1.00E+04 |
| 11 | 1.35E-22 | 1.00E+04 | 4.56E-18 | 1.00E+04 |
| 12 | 7.31E-18 | 1.00E+04 | 2.44E-13 | 1.00E+04 |
| 13 | 6.02E-23 | 1.00E+04 | 2.05E-18 | 1.00E+04 |
| 14 | 2.67E-24 | 1.00E+04 | 9.09E-20 | 1.00E+04 |
| 15 | 1.91E-21 | 1.00E+04 | 6.67E-17 | 1.00E+04 |
| 16 | 2.74E-24 | 1.00E+04 | 9.34E-20 | 1.00E+04 |
| 17 | 4.37E-24 | 1.00E+04 | 1.50E-19 | 1.00E+04 |
| 18 | 9.97E-21 | 1.00E+04 | 3.49E-16 | 1.00E+04 |
| 19 | 1.29E-05 | 6.99E+03 | 4.64E-01 | 7.00E+03 |

**Table C-2. H-3 predicted peak fractional flux to the water table and peak concentration at the 100 meter compliance well for Key Radionuclides scenario runs.**

| Scenario Run | Peak Fractional Flux (mole/yr/mole) | Peak Time (years) | Peak Concentration (pCi/L/Ci) | Peak Time (years) |
|---|---|---|---|---|
| 1 | 4.03E-13 | 1.20E+02 | 1.11E-08 | 1.25E+02 |
| 2 | 1.26E-14 | 1.52E+02 | 3.56E-10 | 1.56E+02 |
| 3 | 7.75E-15 | 1.69E+02 | 2.18E-10 | 1.74E+02 |
| 4 | 3.96E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 5 | 3.96E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 6 | 3.96E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 7 | 3.95E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 8 | 3.96E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 9 | 3.96E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 10 | 3.96E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 11 | 3.97E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 12 | 7.68E-15 | 1.70E+02 | 2.16E-10 | 1.75E+02 |
| 13 | 3.95E-13 | 1.20E+02 | 1.07E-08 | 1.25E+02 |
| 14 | 3.96E-13 | 1.20E+02 | 1.06E-08 | 1.25E+02 |
| 15 | 8.53E-10 | 1.18E+02 | 2.28E-05 | 1.23E+02 |
| 16 | 8.06E-14 | 1.20E+02 | 2.16E-09 | 1.25E+02 |
| 17 | 6.52E-13 | 1.20E+02 | 1.75E-08 | 1.25E+02 |
| 18 | 2.95E-09 | 1.18E+02 | 7.90E-05 | 1.23E+02 |

**Table C-3. I-129 predicted peak fractional flux to the water table and peak concentration at the 100 meter compliance well for Key Radionuclides scenario runs.**

| Scenario Run | Peak Fractional Flux (mole/yr/mole) | Peak Time (years) | Peak Concentration (pCi/L/Ci) | Peak Time (years) |
|---|---|---|---|---|
| 1 | 1.29E-07 | 1.00E+04 | 4.62E-03 | 1.00E+04 |
| 2 | 1.11E-08 | 1.00E+04 | 3.96E-04 | 1.00E+04 |
| 3 | 4.10E-06 | 1.00E+04 | 1.46E-01 | 1.00E+04 |
| 4 | 8.49E-09 | 1.00E+04 | 3.04E-04 | 1.00E+04 |
| 5 | 1.25E-07 | 1.00E+04 | 4.50E-03 | 1.00E+04 |
| 6 | 8.20E-09 | 1.00E+04 | 2.94E-04 | 1.00E+04 |
| 7 | 1.27E-07 | 1.00E+04 | 4.56E-03 | 1.00E+04 |
| 8 | 1.75E-08 | 1.00E+04 | 6.28E-04 | 1.00E+04 |
| 9 | 5.61E-06 | 1.00E+04 | 2.00E-01 | 1.00E+04 |
| 10 | 1.73E-08 | 1.00E+04 | 6.21E-04 | 1.00E+04 |
| 11 | 5.87E-06 | 1.00E+04 | 2.10E-01 | 1.00E+04 |
| 12 | 1.46E-04 | 7.90E+03 | 5.28E+00 | 7.92E+03 |
| 13 | 3.20E-06 | 1.00E+04 | 1.14E-01 | 1.00E+04 |
| 14 | 9.28E-08 | 1.00E+04 | 3.31E-03 | 1.00E+04 |
| 15 | 1.60E-06 | 1.00E+04 | 5.76E-02 | 1.00E+04 |
| 16 | 1.02E-07 | 1.00E+04 | 3.63E-03 | 1.00E+04 |
| 17 | 1.66E-07 | 1.00E+04 | 5.94E-03 | 1.00E+04 |
| 18 | 3.17E-06 | 1.00E+04 | 1.14E-01 | 1.00E+04 |
| 19 | 3.24E-05 | 9.80E+03 | 1.17E+00 | 9.80E+03 |

**Table C-4. Se-79 predicted peak fractional flux to the water table and peak concentration at the 100 meter compliance well for Key Radionuclides scenario runs.**

| Scenario Run | Peak Fractional Flux (mole/yr/mole) | Peak Time (years) | Peak Concentration (pCi/L/Ci) | Peak Time (years) |
|---|---|---|---|---|
| 1 | 7.11E-07 | 1.00E+04 | 1.83E-02 | 1.00E+04 |
| 2 | 2.46E-07 | 1.00E+04 | 7.92E-03 | 1.00E+04 |
| 3 | 2.90E-06 | 1.00E+04 | 7.33E-02 | 1.00E+04 |
| 4 | 4.07E-07 | 1.00E+04 | 1.24E-02 | 1.00E+04 |
| 5 | 6.16E-07 | 1.00E+04 | 1.95E-02 | 1.00E+04 |
| 6 | 4.07E-07 | 1.00E+04 | 1.26E-02 | 1.00E+04 |
| 7 | 6.13E-07 | 1.00E+04 | 1.93E-02 | 1.00E+04 |
| 8 | 4.60E-07 | 1.00E+04 | 1.48E-02 | 1.00E+04 |
| 9 | 2.12E-06 | 1.00E+04 | 4.84E-02 | 1.00E+04 |
| 10 | 4.61E-07 | 1.00E+04 | 1.49E-02 | 1.00E+04 |
| 11 | 2.16E-06 | 1.00E+04 | 4.99E-02 | 1.00E+04 |
| 12 | 1.64E-05 | 1.00E+04 | 3.96E-01 | 1.00E+04 |
| 13 | 1.88E-06 | 1.00E+04 | 4.52E-02 | 1.00E+04 |
| 14 | 5.70E-07 | 1.00E+04 | 1.31E-02 | 1.00E+04 |
| 15 | 1.64E-06 | 1.80E+03 | 5.61E-02 | 3.52E+03 |
| 16 | 6.21E-07 | 1.00E+04 | 1.49E-02 | 1.00E+04 |
| 17 | 8.89E-07 | 1.00E+04 | 2.41E-02 | 1.00E+04 |
| 18 | 6.90E-06 | 3.68E+03 | 2.44E-01 | 4.79E+03 |
| 19 | 3.22E-05 | 9.70E+03 | 1.16E+00 | 9.71E+03 |

**Table C-5. Tc-99 predicted peak fractional flux to the water table and peak concentration at the 100 meter compliance well for technetium scenario runs.**

| Scenario Run | Peak Fractional Flux (mole/yr/mole) | Peak Time (years) | Peak Concentration (pCi/L/Ci) | Peak Time (years) |
|---|---|---|---|---|
| 1 | 5.61E-20 | 1.00E+04 | 2.02E-15 | 1.00E+04 |
| 2 | 1.10E-06 | 1.00E+04 | 3.98E-02 | 1.00E+04 |
| 3 | 3.13E-05 | 9.50E+03 | 1.13E+00 | 9.52E+03 |

**Figure C-1. Infiltration rate through the Upper GCL for three different land use scenarios**

Each curve in Figure C-1 represents a series of time period segments where the infiltration rate for the time period has been averaged.

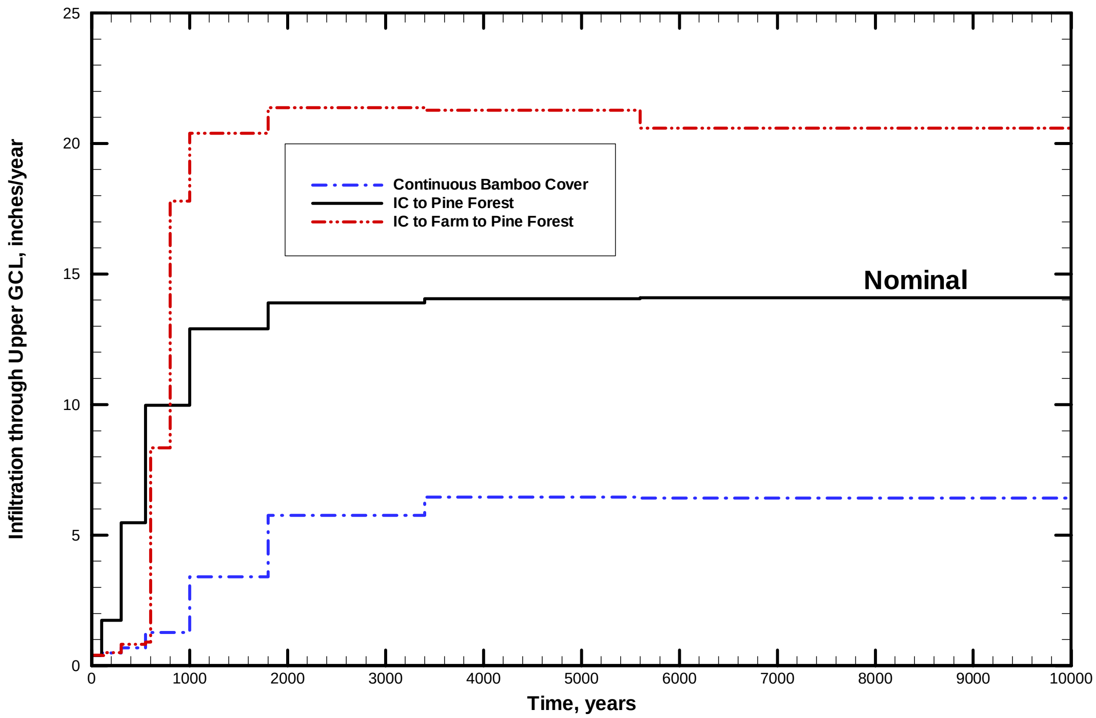


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**Figure C-2. Saturated horizontal hydraulic conductivity of the lower drainage layer for three different land use scenarios.**

The variation over time of the saturated horizontal conductivity of the lower drainage layer.

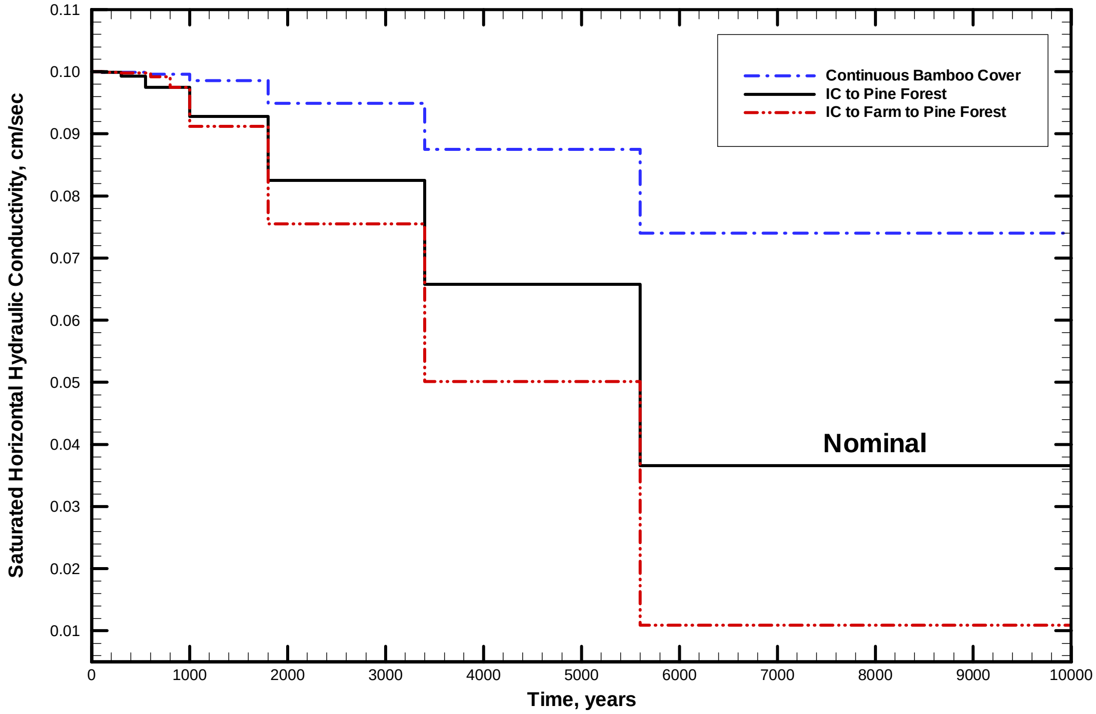


**Figure C-3. Saturated vertical hydraulic conductivity of the lower drainage layer for three different land use scenarios.**

The variation over time of the vault base drainage layer

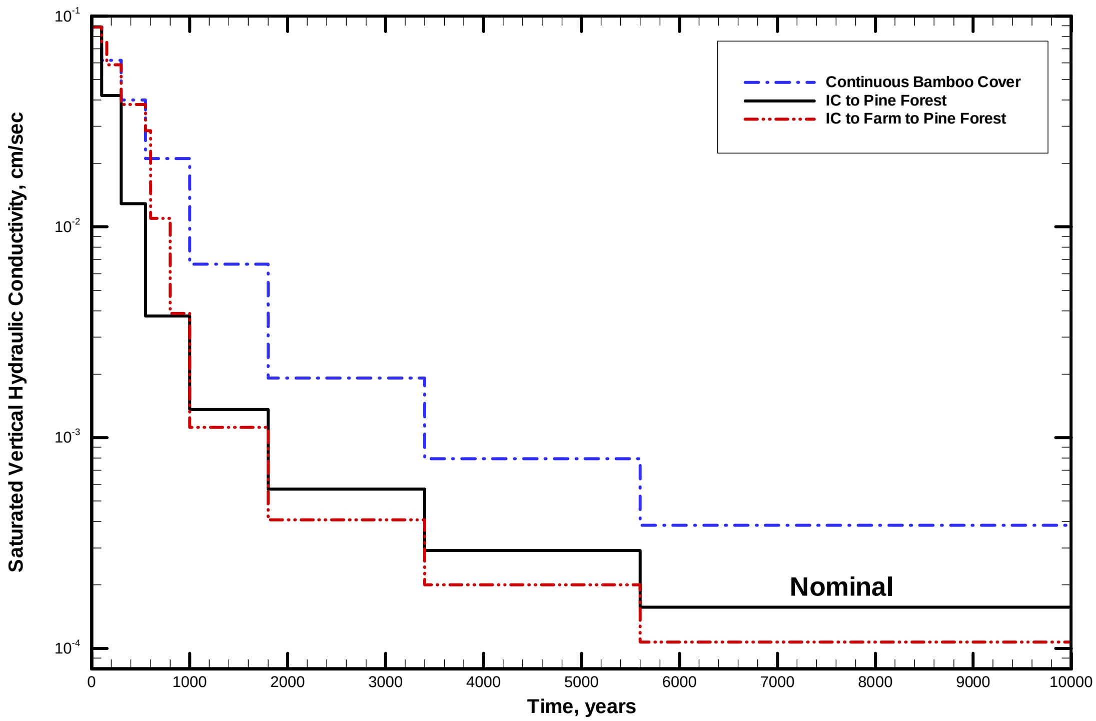


**Figure C-4. Saturated horizontal hydraulic conductivity of the vault base drainage layer for three different land use scenarios**

The variation over time of the saturated horizontal hydraulic conductivity of the vault base drainage layer is shown.

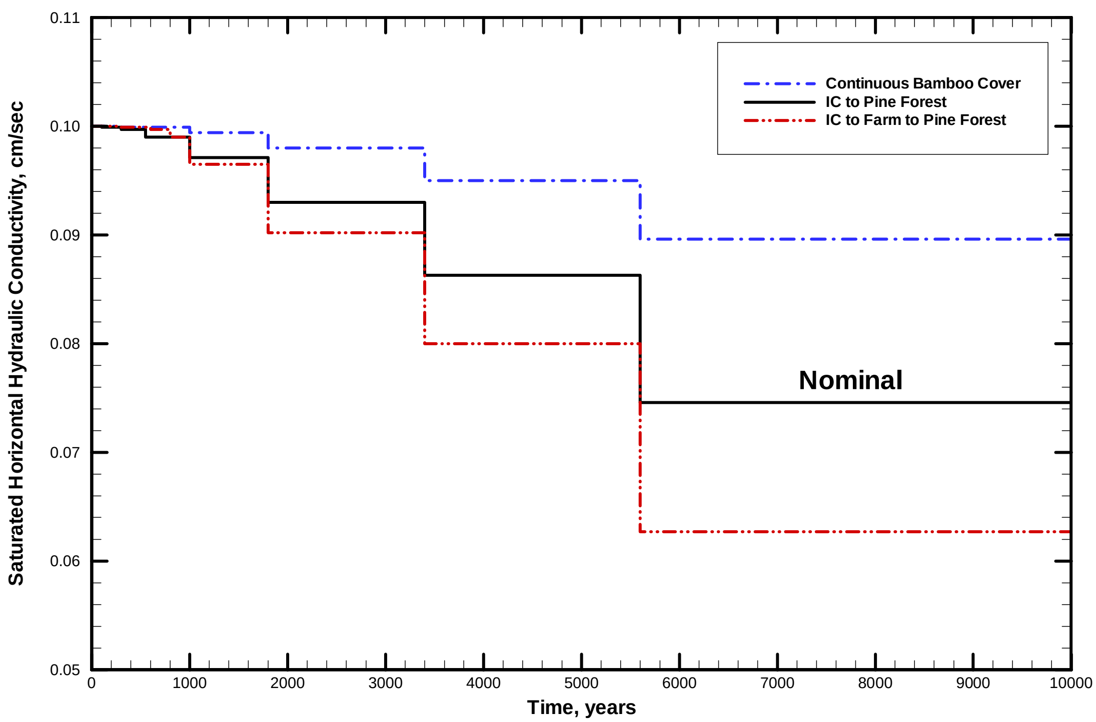


**Figure C-5. Saturated vertical hydraulic conductivity of the vault base drainage layer for three different land use scenarios.**

The variation over time of the saturated vertical hydraulic conductivity of the vault base drainage layer is shown.

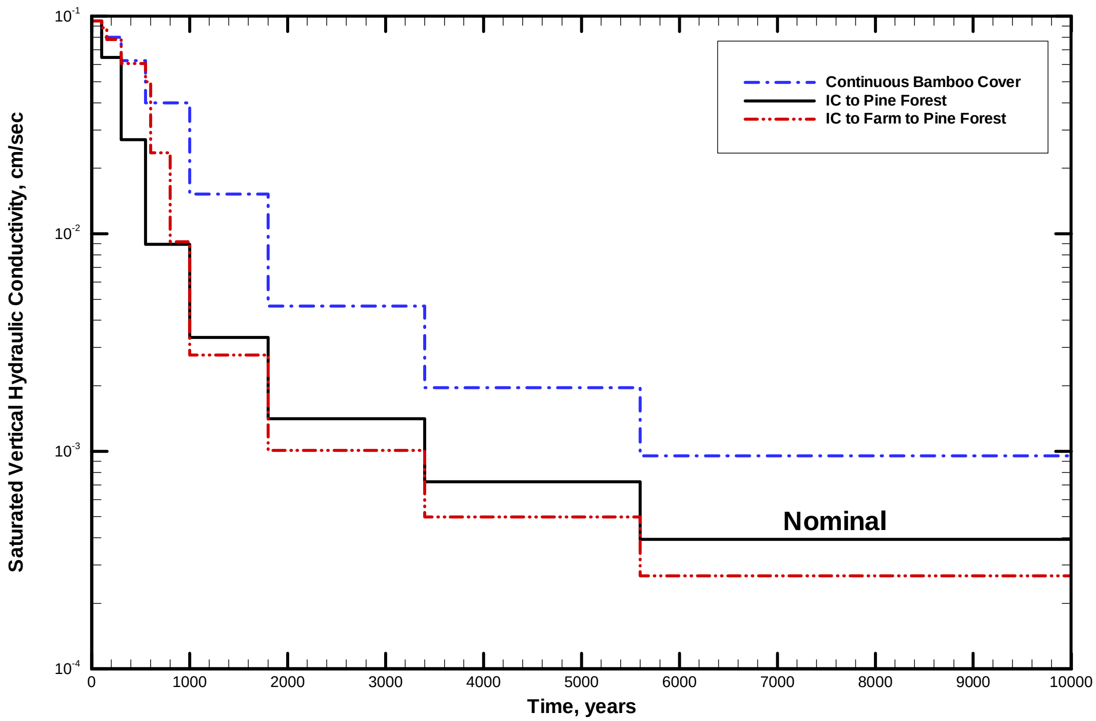


**Figure C-6. Saturated hydraulic conductivity of the vault concrete for three different degradation scenarios.**

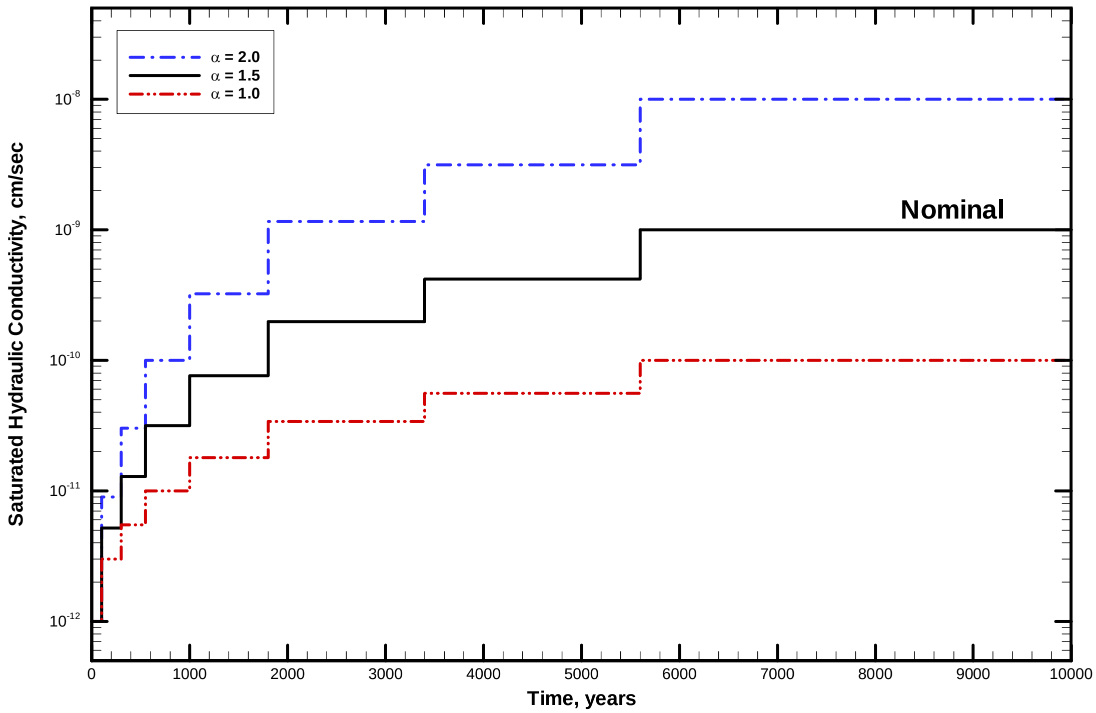


**Figure C-7. Saturated hydraulic conductivity of the vault concrete for an order of magnitude change about nominal conditions.**

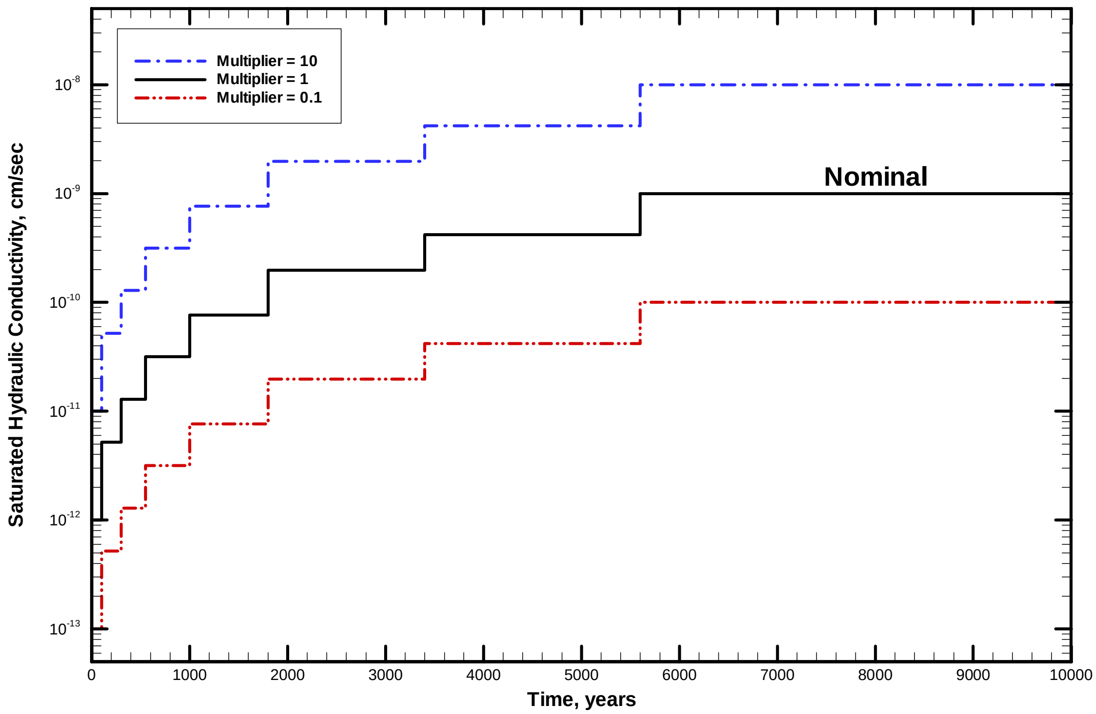


**Figure C-8. Saturated hydraulic conductivity of the Saltstone for three different degradation scenarios.**

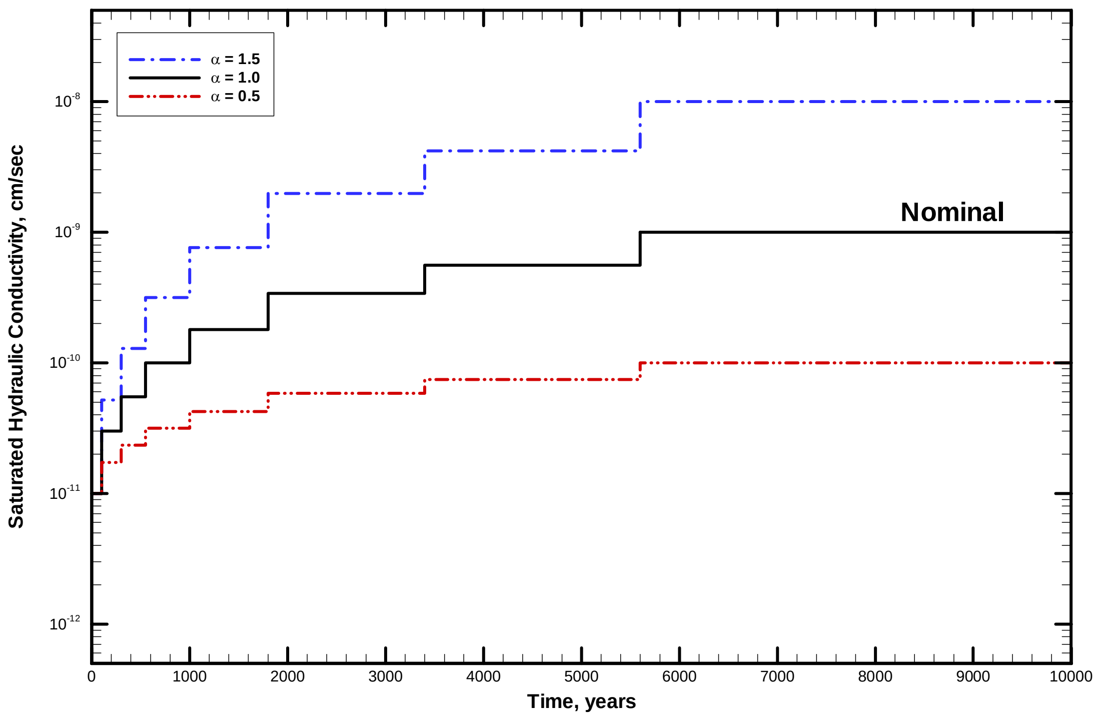
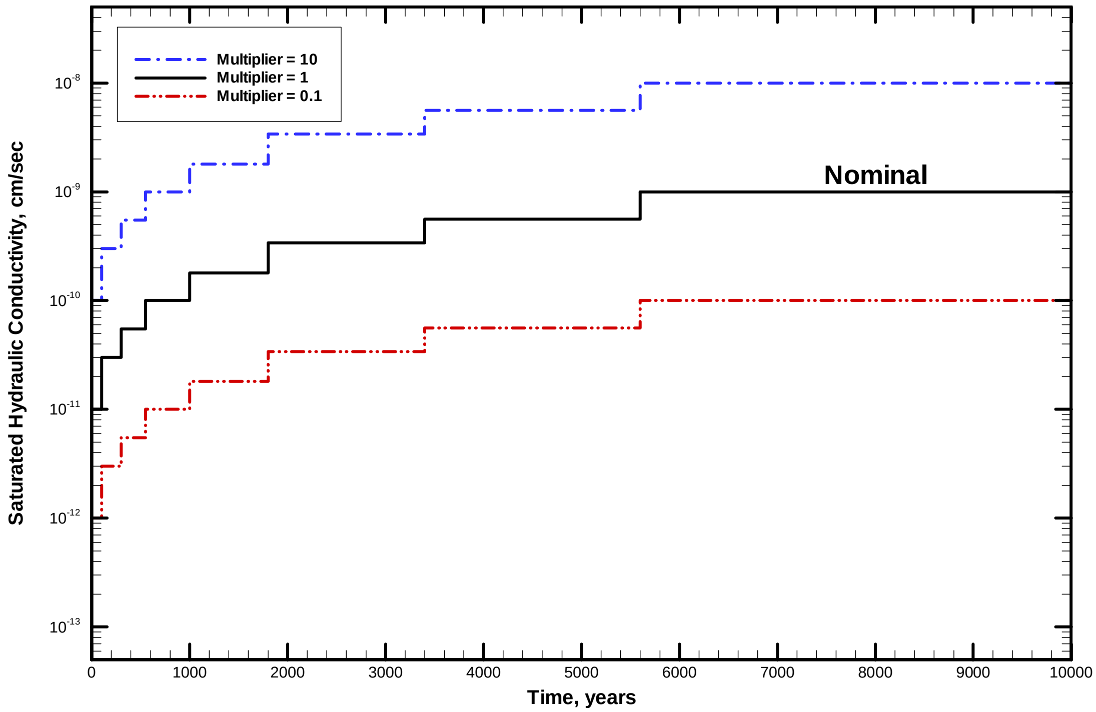


**Figure C-9. Saturated hydraulic conductivity of the Saltstone for an order of magnitude change about nominal conditions.**



# Appendix D. Software Quality Assurance for PORFLOW and Hydrological Evaluation of Landfill Performance (HELP)

Federal rule 10 CFR 830.120, Subpart A establishes quality requirements for Department of Energy (DOE) contractors conducting activities, including providing items and services that affect, or may affect, nuclear safety of DOE facilities. The Department has also developed DOE Order 414.1B, "Quality Assurance" and its associated manuals to ensure quality assurance for all products and services provided by DOE and its contractors. DOE contractors are required to via the S/RID process to identify and incorporate the requirements of 10 CFR 830 and DOE Order 414.1B in their company-level procedures and processes. At SRS, DOE-Savannah River has developed a Quality Assurance Program Manual (SRM 414.1.1.C) which describes its quality assurance program as required by DOE Order 414.1B. The commercial consensus standard upon which the DOE-Savannah River QAP is primarily based is ASME NQA-1-2000, "Quality Assurance Requirements for Nuclear Facility Applications."

The information below describes the Quality Assurance Program implemented by Westinghouse Savannah River Company (WSRC), DOE's operating contractor at the Savannah River Site. This information also describes the software quality assurance plan and test case results for PORFLOW and the software quality assurance for the HELP model, an additional software code used in the performance assessment.

## Software Quality Assurance Requirements

General WSRC requirements for software quality assurance are described in *1Q Quality Assurance Manual*, page 5. The hierarchy of documents is described in *1Q Quality Assurance Manual* as follows:

### 1. WSRC-1-01, Management Policies, MP 4.2, "Quality Assurance"

MP 4.2 contains the WSRC President's policy statement regarding the Company's commitment to provide products and services which meet or exceed the requirements and expectations of our customers. The WSRC Quality Assurance Program is to be implemented in a manner to support implementation of WSRC's imperatives of safety, disciplined operations, cost effectiveness, continuous improvement, and teamwork. WSRC has established and implemented an Integrated Safety Management System (ISMS). The quality assurance (QA) program is consistent with and an integral part of the WSRC ISMS. The policy requires that the program include appropriate procedures to comply with legal, regulatory, contractual, and corporate requirements related to quality. The policy also requires that the WSRC QA program comply with DOE

O 414.1B, 10 CFR 830, Subpart A and the WSRC QA Management Plan. The QA Program applies in a manner which contributes to the safe, reliable, and environmentally sound operation of the SRS. It incorporates a graded approach commensurate with risk in the definition and application of Quality Assurance/Quality Control (QA/QC) requirements. The QA Program provides for the prevention of errors as well as the detection and correction of deficient conditions and incorporates an assessment process for identifying opportunities for continuous improvement. The focus of quality improvement is to reduce the variability of every process that influences the quality and value of the WSRC's products or services.

**2. WSRC-RP-92-225, "Quality Assurance Management Plan"**

The WSRC Quality Assurance Management Plan (QAMP) describes the requirements and responsibilities for execution of the WSRC QA Program for implementing DOE O 414.1B and 10 CFR 830 Subpart A. American Society of Mechanical Engineers Nuclear Quality Assurance (ASME NQA)-1, "Quality Assurance Requirements for Nuclear Facilities" and other consensus standards are used in the development of the WSRC QA Program. The plan has been jointly approved by WSRC and DOE-SR and serves as the basis for the establishment of the procedures contained in this manual.

**3. Procedure Manual 1Q, Quality Assurance Manual**

This manual provides the structure and procedures for achieving and verifying the WSRC requirements for quality. The manual consists of a series of Quality Assurance Procedures (QAPs) which describe applicable quality assurance requirements.

Furthermore, *1Q Quality Assurance Manual*, page 4 states:

> The WSRC QA Program has been developed to be responsive to the requirements of DOE O 414.1B, Quality Assurance and DOE Safety Rule Title 10 CFR 830 Subpart A, Quality Assurance Requirements. Because of the size and complexity of the Savannah River Site (SRS) and its varied products, services, and missions, the program has been defined in a standard framework of company policy, procedures, and instructions to be used by the implementing organizations to perform quality-related activities. These documents shall, as a minimum, include all of the requirements of WSRC-RP-92-225, "WSRC Quality Assurance Management Plan (QAMP)" criteria for which the implementing organizations have responsibility.

In addition to the standard requirements to incorporate DOE O 414.1B in software programs, DOE M 435.1-1 *Radioactive Waste Management Manual,* specifically requires its incorporation as stated on page I-3:

> “(12) **Quality Assurance Program**. Radioactive waste management facilities, operations, and activities shall develop and maintain a quality assurance program that meets the requirements of 10 CFR 830.120, *Quality Assurance Requirements*, and DOE O 414.1, *Quality Assurance*, as applicable.

*1Q Quality Assurance Manual* implements all the requirements stated above. A software quality assurance plan for Porflow was developed to satisfy the Software Quality Assurance Procedure 20-1 of the *1Q Quality Assurance Manual*.

**PORFLOW Software Quality Assurance Plan Description**

The PORFLOW Software Quality Assurance Plan (Collard, 2002) presents the software controls to be applied to PORFLOW. The plan also includes the results of the software grading and the testing and acceptance results. A description of applicable verification and benchmark test cases starts on page 26 of the plan (see appendix).

The plan relies on the “validation” test cases described in ACRi, Inc. 1994. The test cases provide comparisons with published analytical results and with benchmark cases commonly used by similar computer programs. The pertinent and applicable test cases are described and discussed in the plan. Because the results from ACRi, Inc. 1994 were analyzed using an earlier version of PORFLOW, modifications in input files were required by the vendor. In some cases, the models described by the modified input files for the new PORFLOW version did not exactly correspond with the models described by the original input files for the earlier PORFLOW version. A typical example is that the new version moves the boundary nodes from outside the physical model to the edge of the physical model, and in some cases this adjustment was not correctly implemented. Discrepancies were discussed in the plan and were assessed as being insignificant.

**Supporting Qualitative Evidence for Quality Assurance**

While not directly included in the plan, widespread usage of the program and peer review provides additional confidence that the program works properly. The on-line PORFLOW user’s manual (ACRi, Inc. 2005) attests to the usage and testing, where it states:

> **PORFLOWTM** is also distinguished from other computer models by the diversity of its users. Commercial, research and educational organizations in 15 countries are using the software. Among its users are: U.S. DOE, USGS, U.S.NRC, U.S.Army, Southwest Research Institute, Idaho National Engineering Laboratory, Oak Ridge National Laboratory, Savannah River Laboratory, Battelle Pacific Northwest Laboratory, ANDRA (France), SCK-

CEN (Belgium), AECL (Canada), Westinghouse, Lockheed Martin, Fluor Daniel, Rockwell, and a large number of other commercial organizations. Over 100 publications and project reports on the benchmarking, verification and application of **PORFLOWTM** are currently available.

**PORFLOWTM** has been extensively peer-reviewed. Idaho National Engineering Laboratory, Battelle Pacific Northwest, and Prof. Allan Freeze of the University of British Columbia have formally reviewed **PORFLOWTM** or its derivatives. Additionally, it has been reviewed by ANDRA (France), BAe-SEMA (United Kingdom), British Petroleum (United Kingdom), Exxon Production Research, Failure Analysis Associates Inc., Fluor Daniel Inc., Gaz de France (France), SAIC, Shell Oil, SOHIO, and Westinghouse Hanford Company.

The analyses in the 1992 Saltstone PA (reference 1 in the NRC RAI) have been supplemented and those in the 2002 Special Analysis (SA) (reference 3 in the NRC RAI) have been superseded by the Vault 4 SA (Cook et al., 2005). The NRC reviewer noted correctly that the PORFLOW runs performed in support of the 1992 Saltstone PA show a lack of convergence. It has since been determined that this lack of convergence was caused by the use of too large of time increments. In the 2005 Vault 4 SA small grids and time increments were utilized to assure convergence.

**HELP Software Quality Assurance**

The Hydrologic Evaluation of Landfill Performance (HELP) model is a quasi-two-dimensional water balance model designed to conduct landfill water balance analyses. The model requires the input of weather, soil, and design data. It provides estimates of runoff, evapotranspiration, lateral drainage, vertical percolation, hydraulic head, and water storage for the evaluation of various landfill designs. Personnel at the U.S. Army Engineer Waterways Experiment Station in Vicksburg, Mississippi developed the HELP model, under an interagency agreement with the United States Environmental Protection Agency (USEPA). HELP model version 3.07, issued on November 1, 1997, is the latest version of the model available from the Waterways Experiment Station. Documentation for the HELP model is provided in the following USEPA documents:

- USEPA 1994a. *The Hydrologic Evaluation of Landfill Performance (HELP) Model User's Guide for Version 3*, EPA/600/R-94/168a, Office of Research and Development, United States Environmental Protection Agency, Washington, DC. September 1994.
- USEPA (U.S. Environmental Protection Agency). 1994b. *The Hydrologic Evaluation of Landfill Performance (HELP) Engineering Documentation for*

*Version 3*, EPA/600/R-94/168b, Office of Research and Development, United States Environmental Protection Agency, Washington, DC. September 1994.

USEPA 1994b provides the assumptions and limitations associated with the HELP model. A substantial effort was made to provide verification of HELP model version 1.0 which has been documented within the following two USEPA documents:

- USEPA (U.S. Environmental Protection Agency). 1987a. *Verification of the Hydrologic Evaluation of Landfill Performance (HELP) Model Using Field Data*, EPA/600/2-87/050, Office of Research and Development, United States Environmental Protection Agency, Cincinnati, Ohio. July 1987.
- USEPA (U.S. Environmental Protection Agency). 1987b. *Verification of the Lateral Drainage Component of the HELP Model Using Physical Models*, EPA/600/2-87/049, Office of Research and Development, United States Environmental Protection Agency, Cincinnati, Ohio. July 1987.

Within USEPA 1987a, the following was concluded from the verification performed:

> "Simulations of 20 landfill cells from seven sites were performed using the Hydrologic Evaluation of Landfill Performance (HELP) model. Results were compared with field data to verify the model and to identify shortcomings. … The field measurements of the various water budget components varied greatly from cell to cell despite some having identical designs. Consequently, the precision of the verification effort is fairly low, but the study demonstrates that the HELP model is a useful tool for realistically estimating landfill water budgets. Simulation results generally fell within the range of field observations."
>
> "A sensitivity analysis of the HELP model was performed to examine the effects of the major design parameters on components of the water budget for landfills. Hydraulic conductivity values for the topsoil, lateral drainage layers, and clay liners are the most important parameters in determining the water budget components. These parameters are particularly important in estimating the percolation through the landfill."

Based upon this verification modifications to the model have been made to improve predictions. Version 3.07 of the HELP model, issued on November 1, 1997, is the most current version. Based upon this extensive HELP model documentation and verification, it has been accepted by the USEPA and the regulated community as an appropriate water balance model for the examination of landfill designs.

## DESCRIPTION OF APPLICABLE VERIFICATION AND BENCHMARK TEST CASES

Each test case that directly affects the use of PORFLOW at SRS is described below. Of special interest are those test cases that include the flow of water in the vadose zone and the aquifer and the transport of contaminants by diffusion and advection.

Given the above-stated PORFLOW changes and the requisite changes in the input files, each applicable verification and benchmark test case will be described in further detail.

## Verification Test Cases

The verification cases are generally simple and can be compared to analytic solutions.

**Verification Test Case 3**

Verification Test Case 3 examines the solution for transient drawdown. On the GRID command line the descriptor NODES was added. This ensures that the numbers on the command are interpreted as nodes rather than corners and provides consistency between old and new versions of PORFLOW. This descriptor appears throughout most of the test cases and will not be discussed further.

The first node was moved from 0.0 to 0.25 (halfway between the original 0.0 and 0.50) but the last node at 2000 was not moved to 1900.0 (halfway between the original 1800.0 and 2000.0). This likely caused little change in the results and it is unknown which, if either set of input is accurate.

On the BOUNDARY command line for Y, the "-2" was replaced by "Y-" as required. The DIAGNOSTIC and OUTPUT commands were changed with no impact on actual results, because the key information was saved in the archive file, "V3.ARC."

**Verification Test Case5**

Verification Test Case 5 involves coupled flow and heat transfer in a regional flow system. While isothermal models are typically executed at SRS, results from a nonisothermal case that involves flow is applicable in that it demonstrates that the flow portion operates correctly.

For this test case the number of nodes was increased from 41 by 41 to 42 by 42. Rather than specifying the location for each node, the RANGE command was used as a substitute. These develop an identical model, except that in the second case the mesh is finer. Of possible concern would be the location of sources, however, only boundary conditions are applied. The boundary conditions changed according to the convention of "-1" changing to "X-", etc. The nonzero gradient for temperature at the lower Y boundary correctly switched signs. Finally, some of the output specifications were modified.

**Verification Test Case 6**

Verification Test Case 6 involves three-dimensional transport of a contaminant, which is very important to SRS modeling. It consists of a homogeneous, isotropic medium with an infinite horizontal source on the upper surface and a constant horizontal flow. This case very closely mimics most aquifer cases developed at SRS, except that the more complex subsurface consisting of multiple material types is lacking.

For this case, both sets of input coordinates are consistent, but are slightly incorrect. The extent of the X-direction is described as being 3700 m long. The coordinates ranged from –700 to 3000 in both cases. However, they are node locations by default for the older PORFLOW version and are node locations in the newer PORFLOW version by the NODE descriptor (corners are the default), rather than the desired corner locations.

Similarly the extent of the Y-direction is described as being 800 m long. The original node coordinates ranged from –10 to 800. This placed the lowest node in the range at the correct location because the corner of the physical model would be at zero, halfway between –10 and +10 for the first and second nodes. However the upper corner would be at 745, halfway between the 800 and the 690 of the next to highest node in the range. The more recent data set extends from 0 to 800 but it specifically calls out the data as nodes, which is incorrect because it should have been corner data.

The extent of the Z-direction is described as 56 m. The original data ranged from -56 to 0.05. Only the upper node is at the correct location because the corner would be at zero, halfway between –0.05 and +0.05. The new data ranges from –50 to 0 as nodes. This is incorrect because the lower location has been changed from –56 to –50.

The boundary conditions are set to zero flux at all boundaries except the lower X boundary where the concentration is set to zero. This caused the input line to be changed from "-1" to "X-". However, PORFLOW changed the definition of the flux condition on a boundary command. Originally the flux option meant that advection could still move contaminants across the boundary, but in the more recent PORFLOW versions, even this is prevented. Typically contaminants are transported to a boundary but cannot penetrate it, thus they rapidly accumulate at the boundary. If only results in the interior of the model are important, then this effect is minor only affecting the mass balance.

The geometric property was omitted in the newer version, thus the calculation of the properties of the host porous matrix at the element interface would default to the harmonic mean.

Integration of the concentration by the CONDIF approach was omitted in the newer version. The CONDIF approach as described in Runchal, 1997 is provided below.

“The numerical integration starts with the assumption of an integration profile for the state variable. Two different kinds of profiles are employed. These are the first- and second-order polynomial profiles and the exponential profile. These integration profiles result, respectively, in the ‘upwind’, and the central difference and, the exponential schemes. The first two are schemes combined in a hybrid scheme. The central difference scheme, which provides second-order accuracy, is the preferred scheme. However, use of the central difference scheme may result in numerical instabilities if the magnitude of the local value of the grid Peclet number exceeds 2. With U, $\delta L$ and $\Gamma$, respectively, as the velocity component, grid interval and diffusivity in a given direction, the grid Peclet number, Pe, is defined as:

$$Pe = U \delta L / \Gamma.$$

The local value of the Peclet number at each grid node is constantly monitored in each direction. If $Pe > 2$, then the numerical scheme automatically shifts to the ’upwind’ formulation. This method of enhancing stability is known as the hybrid scheme (Runchal, 1972). The hybrid scheme has second-order accuracy if the $Pe < 2$; otherwise, it is only first-order accurate. Because upwinding results in an increasing amount of numerical diffusion as the angle between the velocity vector and the grid lines increases, PORFLOW$^{TM}$ allows the use of an exponential numerical scheme (Spalding, 1972) to represent the exact solution of the one-dimensional form of transport equations without sources. Th eexponential scheme cannot be accurately classified; however, in practice, it is known to decrease numerical dispersion if the flow is primarily unidirectional and source terms are small. Otherwise, its accuracy is comparable to that of the hybrid scheme. An alternate method to obtain numerical stability with second-order accuracy is that of the CONDIF scheme (Runchal, 1987b) which is a modified central-difference scheme. It is a second-order member of the TVD family of numerical schemes (Harten, 1983) that leads to an unconditionally stable formulation. A third option which is available is that of a version of the QUICK scheme (Leonard, 1979) which has been adapted for nonorthogonal grids.

The user controls the method of evaluation of the integrals, which is equivalent to the selection of a ‘basis function’ in the finite-element technique. For most problems, the hybrid scheme is sufficient. If the grid is very coarse, then the CONDIF or the QUICK scheme should be employed. “

ACRI, 1994 states:

> “The maximum Peclet number for the grid employed is 5.5 and the maximum Courant number is 0.04. Since the Peclet number is almost three times the desired value of 2, some numerical errors may be present. These results could be improved by smaller grid size.”

Personal communication with Runchal indicated that results from the newer PORFLOW version were in close agreement with earlier results. Thus, in spite of removing the CONDIF control that helps compensate for a coarse grid the results were quite reasonable.

The text states that the problem is symmetric in the lateral (y) direction, hence only half the domain was simulated. The text and the figure show a domain of 800 m with the source in the center. If only half the domain in the y direction were modeled, the model would encompass only 400 m, but the input file encompasses 800 m, thus the text and the input file are inconsistent.

No original convergence criteria were specified thus it defaulted to 0.001. The revised convergence was 1.E-7, which is much tighter.

Minor changes to the diagnostics, history and output selections were noted. The solution originally was set to about 1.58E8 seconds in uniform steps of about 3.15E4 seconds. The revision started with steps of 2.E3 seconds that increased to a maximum of 5E6 seconds. These are all subjective. While the magnitude of the Courant number would increase, if the problem has stabilized by the time it becomes large, there should be minimal effect on the final results.

**Verification Test Case 7**

Verification Test Case 7 involves Philip's horizontal unsaturated flow case where a wetting front is initiated by a pressure change at one boundary. Primarily, only minor changes were noted in the GRID command, and adjusting the BOUNDARY command, the DIAGNOSTIC command and the OUTPUT command. The extent in the X-direction should be 20 cm, but because node locations are used the actual extent of the physical model is shortened slightly.

**Verification Test Case 8**

Verification Test Case 8 involves Philip's vertical unsaturated column that is similar to the Philip's horizontal column, but the column is vertical so that capillary and gravity forces can take effect. In both cases the range for the Y coordinate is set to 15 cm. In the original version of PORFLOW, the default was for nodes, which generated a slightly shorted physical domain. In the newer PORFLOW version, the default is for corners, which matches the physical domain with the text. Minor changes are apparent in that the order of some commands has changed, the BOUNDARY input has been modified and the DIAGNOSTIC and OUTPUT commands have been adjusted.

**Verification Test Case 9**

Verification Test Case 9 involves steady-state infiltration from a line source to a water table. This case involves modeling the vadose zone with the water table as its lower boundary, similar to the vadose zone modeling at SRS.

Here the coordinates are specified by the range option. The range option in the original PORFLOW version used corners rather than nodes as the default (contrary to statements in the user's manual). Both data sets for coordinates are correct.

Minor changes to the GRID command, BOUNDARY commands, the DIAGNOSTIC command and the OUTPUT command were noted between the two sets of input files. The boundary condition for the pressure at the Y- face changed from "interface" to "value." The original PORFLOW allowed the user to prescribe a value at the node with "value" or a value at the element interface, i.e., at the edge of the physical model with "interface." Because the newer version of PORFLOW moves the location of the boundary node to the edge of the physical model, the "value" and the "interface" are synonymous and are equivalent to the previous "interface." "Interface" has been omitted from the newer PORFLOW, so older input sets that relied on the "value" may produce different results if used with the newer PORFLOW.

The relation between the pressure and the saturation is expressed as a Brooks & Corey relationship in the original data set, but as an exponential relationship in the subsequent data set.
For a steady-state solution the difference apparently has minimal effect on the final results.

**Verification Test Case 10**
Verification Test Case 10 involves free-surface Boussinesq flow with recharge from one side in a semi-infinite, unconfined aquifer. The extent of the model in the X-direction is 200 m. Both data sets employ a minimum and maximum for the X that apparently properly describes the physical model. However, the earlier version of PORFLOW used a default of nodes, thus the physical model would have been slightly smaller than the defined model. The Y-direction had an extent of 11 m. The original model prescribed nodes that extended from 0 to 11.1. The physical boundaries would have been from 1 to 11, or only 10 m in extent. The more recent data set prescribes nodes from 0 to 11, and because the boundary nodes in the later PORFLOW are aligned with the physical model boundaries this prescription is correct.

Initial conditions were originally prescribed with the INITIAL command. The more recent version uses a combination of the SET command and a BOUNDARY command with the same effect.

The convergence is tightened from 1E-6 to 1E-10 in the later data set, although the maximum number of iterations is reduced from 1000 to 25 producing a tradeoff.

The BOUNDARY command, DIAGNOSTIC command and the OUTPUT command are modified. The DIAGNOSTIC command is misspelled as DIAGNOSITC in both

versions, but PORFLOW only relies on the first four characters, thus the operation of the model will not be affected.

**Verification Test Case 11**

Verification Test Case 11 involves free-surface Boussinesq flow with seepage from the surface in an unconfined aquifer. All the comments for Verification Test Case 10 apply here.

## Benchmark Test Cases

The benchmark test cases produce solutions that are compared to solutions from other computer codes, because typically the cases are too complex to afford analytic solutions. All the test cases "have been used previously for validation of other computer codes" (ACRI, 1994). Having test cases that were important enough to use for validation of other computer codes indicates that they are excellent candidates for the PORFLOW validation.

**Benchmark Test Case 1**

Benchmark Test Case 1 involves two-dimensional transient infiltration. The model size is described as being 15 cm in the X-direction and 10 cm in the Y-direction. The original data set provided a minimum and a maximum for the X-direction as 0 and 0.15 m. Given the default of nodes, the size of the physical domain would be slightly short, by 0.01 m. The Y coordinate was described as a range of 0.1 m, which would be correct. For the later PORFLOW version, because the boundary nodes are aligned with the edge of the physical model, the same BOUNDARY commands with the NODE modifier produce a correct model.

Other minor changes are found in the BOUNDARY command, the DIAGNOSTIC command and the OUTPUT command.

Output results were compared with results from the TOUGH computer program.

**Benchmark Test Case** *2*

Benchmark Test Case 2 involves two-dimensional steady-state infiltration. The model size is 150 m in the X direction and 35 m in the Y direction (the figure shows a Y range of 42 m). The original data set had a range in the X coordinate from 0 to 150 for the nodes. It would have ranged from 2.5 to 147.5 for the physical model or only 145 m, rather than the intended 150 m. The Y coordinate ranged from –0.1 to 36 for the nodes. The range for the physical model would have been 0 to 35, which was correct.

The newer data set defined ranges that were correct. The newer data set increased the number of nodes in the X-direction from 31 to 32. This helped align the corners (or cell faces) at 5 m intervals. The number of nodes in the Y-direction remained at 33. The original data set had distances between node locations that varied from 0.2 m to 2 m to

provide better resolution near the location of the phreatic surface. Specifying a range forces all distances to equal values and that resolution is lost.

The older version specified a datum of 0,0 while none was specified for the newer data set, but because the defaults are zero these are equivalent.

The BOUNDARY commands were changed to reflect the new format. The boundary condition along the upper X boundary was correctly applied by increasing the location from 31 to 32 that matched the number of nodes in the X direction. The boundary condition at Y+ was a flux of –1 in the original (downward) and +1 in the newer data set (into the domain which is downward).

The convergence criterion was loosened from 1E-5 to 1E-3 with a maximum iteration count changed from a default of 100 to 500. The original data set required that the problem be solved for 4 years while the late data set required a steady state solution. As long as a satisfactory solution is achieved, the convergence criterion difference is not important.

The output region for the newer data set was selected for nodes 1,1 to 999,999 with an interval of 2,2. This allowed half of the data to be skipped. PORFLOW allows the 999,999 upper limit even though the size of the problem domain is only 32 by 33.

The results from this test case were compared with FEMWATER results.

**Benchmark Test Case 3**
Benchmark Test Case 3 is a simulation of the Jornada Test Trench in an extremely dry heterogeous soil. The area was heavily instrumented and infiltration experiments were conducted. This case qualifies both as a validation case and as a short-term field validation case. Because the SRS site is a much wetter site the value of the field validation is limited. However, it is much more difficult to simulate extremely dry conditions thus this is a challenging test case.

The extent of the model is 800 cm in the X direction and 650 cm in the Y direction. The original data set had a range of X-direction nodes from –5 to 820 and X-direction corners from 0 to 800 that properly described the physical model. Similarly the range of Y-direction nodes was from -655 to 5 and X-direction corners from –650 to 0. The newer data set took advantage of the boundary nodes' locations at the edge of the physical boundary and placed the lower and upper nodes at the proper locations of 0 and 800 for the X direction and –650 and 0 for the Y direction.

The BOUNDARY commands were modified to conform to the new convention. The flux for the pressure equation at the upper Y boundary was a –2 cm/day in the original data set (downward) while in the newer data set it was +2 cm/day (inward), but it is also downward.

The DIAGNOSTIC command was modified to report less often and its location was moved.

Two convergence criteria were provided in the newer data set, but none in the original data set. One CONVERGENCE command is for FLOW that is not described in either user's manual, although an example is provided in the newer user's manual. The CONVERGENCE command for pressure in the newer data set is equivalent to the default criterion.

This benchmark case was compared to results from the FLASH code and the TRACER3D code.

## CONCLUSIONS OF VALIDATION TESTING

PORFLOW has been tested previously by international experts such as Allan Freeze and by the Southwest Research Center for the Nuclear Regulatory Commission (Collard,1998). The high level of testing by experts and the extensive test cases included, indicates that PORFLOW can be readily applied for SRS problems with the expectation of correct program performance.

Some of the problems for which PORFLOW has been validated are as follow:

- Vadose zone flow problems with multiple soil types
- Vadose zone contaminant transport problems with multiple soil types and simple chemistry
- Aquifer flow problems with multiple soil types
- Aquifer contaminant transport problems with multiple soil types and simple chemistry.

Cases with contaminant sources were documented. Multiphase and nonisothermal cases were not documented.

# APPENDIX D References

ACRi, (Analytical & Conmputational Research, Inc.),1994, *PORFLOW Validation Version 2.50*.

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# APPENDIX E. Input for Potential Water Usage Inside the 100 Meter Buffer Zone Sensitivity

**Table E-1. Predicted Peak Fluxes over 10,000 Years (Cook et al., 2005)**

| Nuclides | Peak Flux mol/yr/mol | Peak Time years |
|---|---|---|
| NO3 | 3.24E-05 | 9.80E+03 |
| Al-26 | 5.49E-13 | 1.00E+04 |
| Am-243 | 1.43E-32 | 1.00E+04 |
| Np-239 | 4.53E-36 | 1.00E+04 |
| Pu-239 | 4.53E-27 | 1.00E+04 |
| Pu5-239 | 1.65E-30 | 1.00E+04 |
| Bi-210 | 0.00E+00 | |
| Po-210 | 0.00E+00 | |
| C-14 | 3.44E-24 | 1.00E+04 |
| Cf-249 | 3.71E-34 | 5.76E+03 |
| Cm-245 | 7.17E-34 | 1.00E+04 |
| Pu-241 | 1.38E-35 | 1.00E+04 |
| Pu5-241 | 5.06E-39 | 1.00E+04 |
| Am-241 | 1.07E-34 | 1.00E+04 |
| Np-237 | 3.82E-24 | 1.00E+04 |
| Cl-36 | 1.88E-23 | 1.00E+04 |
| Cm-245 | 1.24E-38 | 1.00E+04 |
| Pu-241 | 4.48E-40 | 1.00E+04 |
| Pu5-241 | 1.75E-43 | 1.00E+04 |
| Am-241 | 2.32E-37 | 1.00E+04 |
| Np-237 | 3.96E-24 | 1.00E+04 |
| Cm-246 | 6.54E-39 | 1.00E+04 |
| Cm-247 | 2.82E-38 | 1.00E+04 |
| Am-243 | 2.40E-36 | 1.00E+04 |
| Np-239 | 7.62E-40 | 1.00E+04 |
| Pu-239 | 9.20E-31 | 1.00E+04 |
| Pu5-239 | 3.34E-34 | 1.00E+04 |
| Cm-248 | 2.76E-38 | 1.00E+04 |
| Pu-244 | 1.58E-28 | 1.00E+04 |
| Pu5-244 | 5.76E-32 | 1.00E+04 |
| Cs-135 | 1.10E-14 | 1.00E+04 |
| Cs-137 | 1.42E-41 | 1.46E+03 |
| H-3 | 4.03E-13 | 1.20E+02 |
| I-129 | 1.29E-07 | 1.00E+04 |
| K-40 | 6.97E-08 | 1.00E+04 |
| Mo-93 | 8.21E-08 | 1.00E+04 |
| Nb-93m | 6.72E-12 | 1.00E+04 |
| Nb-94 | 3.33E-21 | 1.00E+04 |
| Nb-95m | 0.00E+00 | |
| Nb-95 | 0.00E+00 | |
| Ni-59 | 2.37E-18 | 1.00E+04 |
| Np-237 | 7.25E-24 | 1.00E+04 |
| Pd-107 | 1.25E-16 | 1.00E+04 |

**Table E-1. Predicted Peak Fluxes over 10,000 Years (Cook et al., 2005)**

| Nuclides | Peak Flux mol/yr/mol | Peak Time years |
|---|---|---|
| Pu-238 | 5.59E-42 | 2.60E+03 |
| Pu5-238 | 2.07E-45 | 2.60E+03 |
| U-234 | 4.13E-26 | 1.00E+04 |
| Pu-239 | 7.75E-27 | 1.00E+04 |
| Pu5-239 | 2.81E-30 | 1.00E+04 |
| U-235 | 1.83E-27 | 1.00E+04 |
| Pu-240 | 3.59E-27 | 1.00E+04 |
| Pu5-240 | 1.30E-30 | 1.00E+04 |
| U-236 | 5.85E-27 | 1.00E+04 |
| Pu-241 | 3.93E-68 | 1.06E+03 |
| Pu5-241 | 1.64E-71 | 1.06E+03 |
| Am-241 | 4.00E-39 | 1.00E+04 |
| Np-237 | 7.25E-24 | 1.00E+04 |
| Pu-242 | 1.01E-26 | 1.00E+04 |
| Pu5-242 | 3.68E-30 | 1.00E+04 |
| U-238 | 1.26E-28 | 1.00E+04 |
| Pu-244 | 1.03E-26 | 1.00E+04 |
| Pu5-244 | 3.75E-30 | 1.00E+04 |
| Ra-226 | 5.55E-19 | 1.00E+04 |
| Rb-87 | 2.38E-15 | 1.00E+04 |
| Se-79 | 7.11E-07 | 1.00E+04 |
| Sn-126 | 2.03E-22 | 1.00E+04 |
| Sr-90 | 4.32E-19 | 5.62E+02 |
| Tc-99 | 5.61E-20 | 1.00E+04 |
| Th-228 | 0.00E+00 | |
| Ra-224 | 0.00E+00 | |
| Th-229 | 1.21E-36 | 1.00E+04 |
| Ra-225 | 4.32E-41 | 1.00E+04 |
| Ac-225 | 3.23E-41 | 1.00E+04 |
| Th-230 | 2.85E-36 | 1.00E+04 |
| Ra-226 | 8.04E-21 | 1.00E+04 |
| Pb-210 | 2.16E-22 | 1.00E+04 |
| Po-210 | 6.60E-24 | 1.00E+04 |
| Th-232 | 3.13E-36 | 1.00E+04 |
| Ra-228 | 9.13E-45 | 1.00E+04 |
| Th-228 | 4.74E-46 | 1.00E+04 |
| Ra-224 | 1.59E-47 | 1.00E+04 |
| U-232 | 2.38E-48 | 2.79E+03 |
| Th-228 | 1.66E-50 | 2.80E+03 |
| Ra-224 | 5.58E-52 | 2.80E+03 |
| U-233 | 4.45E-26 | 1.00E+04 |
| Th-229 | 5.04E-29 | 1.00E+04 |
| Ra-225 | 1.79E-33 | 1.00E+04 |
| U-234 | 4.52E-26 | 1.00E+04 |
| Th-230 | 3.58E-29 | 1.00E+04 |
| Ra-226 | 2.86E-23 | 1.00E+04 |
| Pb-210 | 7.72E-25 | 1.00E+04 |
| Po-210 | 2.36E-26 | 1.00E+04 |

**Table E-1. Predicted Peak Fluxes over 10,000 Years (Cook et al., 2005)**

| Nuclides | Peak Flux mol/yr/mol | Peak Time years |
|---|---|---|
| U-235 | 4.65E-26 | 1.00E+04 |
| Pa-231 | 1.09E-30 | 1.00E+04 |
| Ac-227 | 8.86E-34 | 1.00E+04 |
| Th-227 | 2.93E-37 | 1.00E+04 |
| Ra-223 | 1.15E-36 | 1.00E+04 |
| U-236 | 4.65E-26 | 1.00E+04 |
| U-238 | 4.65E-26 | 1.00E+04 |
| Th-234 | 1.72E-37 | 1.00E+04 |
| U-234 | 7.12E-32 | 1.00E+04 |
| Zr-93 | 2.22E-27 | 1.00E+04 |
| Nb-93m | 9.19E-32 | 1.00E+04 |
| Zr-95 | 0.00E+00 | |
| Nb-95 | 0.00E+00 | |

## Table E-2. Source Terms

For each of the contaminants and all daughters, the source terms are expressed as the fractional release to the water table calculated by the unsaturated-zone modeling. The fractional release has the unit of mole/year/mole of parent. The time history of each component is used as the source term. The amount released is assumed to be evenly distributed to the total volume of the 12 source cells listed in Table A-3. Based on the grid coordinates, the volumes of all these cells are calculated (Table A-4). The total volume is $6.1215\times10^5$ ft$^3$

**Table E-2**
**Source Node Locations and Volumes (Cook et al., 2005)**

| I | J | K | XC | YC | ZC | VOL |
|---|---|---|---|---|---|---|
| 13 | 13 | 14 | 21350.0 | 11750.0 | 230.110 | 5.1200E+04 |
| 13 | 14 | 14 | 21350.0 | 11850.0 | 230.650 | 5.0900E+04 |
| 13 | 15 | 14 | 21350.0 | 11950.0 | 231.306 | 5.0525E+04 |
| 14 | 12 | 14 | 21450.0 | 11650.0 | 229.997 | 5.1250E+04 |
| 14 | 13 | 14 | 21450.0 | 11750.0 | 230.353 | 5.1100E+04 |
| 14 | 14 | 14 | 21450.0 | 11850.0 | 230.822 | 5.0850E+04 |
| 14 | 15 | 14 | 21450.0 | 11950.0 | 231.405 | 5.0500E+04 |
| 15 | 10 | 14 | 21550.0 | 11450.0 | 229.486 | 5.1525E+04 |
| 15 | 11 | 14 | 21550.0 | 11550.0 | 229.935 | 5.1250E+04 |
| 15 | 12 | 14 | 21550.0 | 11650.0 | 230.340 | 5.1050E+04 |
| 15 | 13 | 14 | 21550.0 | 11750.0 | 230.699 | 5.0925E+04 |
| 16 | 11 | 14 | 21650.0 | 11550.0 | 230.306 | 5.1075E+04 |
| | | | | | TOTAL | 6.1215E+05 |

The fractional release is divided by the total volume to obtain the concentration increments in the source nodes in mole/ ft$^3$/mole parent. However, because fractional release is often a very small number, within PORFLOW we multiply it by $10^{12}/6.1215\times10^5$ ft$^3$ = $1.6336\times10^6$. The concentration unit in PORFLOW saturated-zone computation is, therefore, pico-mole/ft$^3$/mole parent. This multiplication factor is the same for every contaminant. PORFLOW has a "SCALE" command so that users can apply it to each fractional release time history. In PORFLOW 5.97.0, the scaling is performed by the code if a user enters "TOTAl VOLUme" in the SOURce command. The source terms are read by a PORFLOW input file.

The flux terms exiting the bottom of the unsaturated zone model was processed using a Fortran program to truncate the fluxes less than $10^{-20}$ times the peak flux such that only the significant part of the output flux profile was utilized to generate the input source terms for the saturated zone model

**Table E-3 (Cook et al., 2005)**
**Node Indices for Locations of the Source Nodes**

| I | J | K |
|---|---|---|
| -- | -- | -- |
| 13 | 13 | 14 |
| 13 | 14 | 14 |
| 13 | 15 | 14 |
| 14 | 12 | 14 |
| 14 | 13 | 14 |
| 14 | 14 | 14 |
| 14 | 15 | 14 |
| 15 | 10 | 14 |
| 15 | 11 | 14 |
| 15 | 12 | 14 |
| 15 | 13 | 14 |
| 16 | 11 | 14 |

**Table E-4**
**Source Node Locations and Volumes (Cook et al., 2005)**

| I | J | K | XC | YC | ZC | VOL |
|---|---|---|---|---|---|---|
| -- | -- | -- | ------- | ------- | ------- | ---------- |
| 13 | 13 | 14 | 21350.0 | 11750.0 | 230.110 | 5.1200E+04 |
| 13 | 14 | 14 | 21350.0 | 11850.0 | 230.650 | 5.0900E+04 |
| 13 | 15 | 14 | 21350.0 | 11950.0 | 231.306 | 5.0525E+04 |
| 14 | 12 | 14 | 21450.0 | 11650.0 | 229.997 | 5.1250E+04 |
| 14 | 13 | 14 | 21450.0 | 11750.0 | 230.353 | 5.1100E+04 |
| 14 | 14 | 14 | 21450.0 | 11850.0 | 230.822 | 5.0850E+04 |
| 14 | 15 | 14 | 21450.0 | 11950.0 | 231.405 | 5.0500E+04 |
| 15 | 10 | 14 | 21550.0 | 11450.0 | 229.486 | 5.1525E+04 |
| 15 | 11 | 14 | 21550.0 | 11550.0 | 229.935 | 5.1250E+04 |
| 15 | 12 | 14 | 21550.0 | 11650.0 | 230.340 | 5.1050E+04 |
| 15 | 13 | 14 | 21550.0 | 11750.0 | 230.699 | 5.0925E+04 |
| 16 | 11 | 14 | 21650.0 | 11550.0 | 230.306 | 5.1075E+04 |
| | | | | | ------- | ---------- |
| | | | | | TOTAL | 6.1215E+05 |