December 28, 2005

Mr. Charles E. Anderson
Principal Deputy Assistant Secretary
Office of Environmental Management
U.S. Department of Energy
1000 Independence Avenue, S.W.
Washington, DC 20585

SUBJECT: TECHNICAL EVALUATION REPORT FOR DRAFT WASTE DETERMINATION FOR SALT WASTE DISPOSAL

Dear Mr. Anderson:

Under Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA), the Secretary of Energy, in consultation with the U.S. Nuclear Regulatory Commission (NRC), may determine that certain radioactive waste resulting from reprocessing of spent nuclear fuel is not high-level waste. By letter dated March 31, 2005, the U.S. Department of Energy (DOE) submitted the "Draft Section 3116 Determination, Salt Waste Disposal, Savannah River Site" to the NRC for review. The NRC staff has performed a technical review to determine whether there is reasonable assurance that applicable criteria of the NDAA can be met by the waste management approach proposed in DOE's draft Section 3116 waste determination. The NRC's assumptions, analysis, and conclusions are presented in the attached Technical Evaluation Report (TER).

Based on the information provided by DOE to the NRC in letters dated March 31, June 30, July 15, September 15, and September 30, 2005, the NRC staff has concluded that there is reasonable assurance that the applicable criteria of the NDAA can be met provided certain assumptions made in DOE's analyses are verified via monitoring. The assumptions described in Section 4.3.1 of the TER are important to demonstrating that the performance objectives in 10 CFR 61, Subpart C, can be met and fall into the following general categories: wasteform and vault degradation, the effectiveness of infiltration and erosion controls, and estimation of the radiological inventory. The NDAA requires NRC, in coordination with the State of South Carolina, to monitor disposal actions taken by DOE for the purpose of assessing compliance with the performance objectives of 10 CFR 61, Subpart C. Consequently, NRC requests that DOE develop proposed approaches that DOE will use to address the areas identified in Section 4.3.1. NRC will then, in coordination with the State, develop a program by which NRC will monitor DOE's implementation of the approaches.

It is important to note that the NRC's conclusions presented in this TER are based on the information provided by DOE. If, in the future, DOE determines it is necessary to revise its assumptions, analysis, design, or waste management approach and those changes are important to meeting the criteria of the NDAA, DOE should consult once again with NRC regarding the enclosed TER. The NRC looks forward to continuing to work cooperatively with DOE in implementing the Section 3116 requirements. If you have any questions or need additional information regarding this TER, please call me at 301-415-7437, or call Anna Bradford, senior project manager on my staff, at 301-415-5228.

Sincerely,

/**RA**/

Larry W. Camper, Director
Division of Waste Management
and Environmental Protection
Office of Nuclear Material Safety
and Safeguards

It is important to note that the NRC's conclusions presented in this TER are based on the information provided by DOE. If, in the future, DOE determines it is necessary to revise its assumptions, analysis, design, or waste management approach and those changes are important to meeting the criteria of the NDAA, DOE should consult once again with NRC regarding the enclosed TER. The NRC looks forward to continuing to work cooperatively with DOE in implementing the Section 3116 requirements. If you have any questions or need additional information regarding this TER, please call me at 301-415-7437, or call Anna Bradford, senior project manager on my staff, at 301-415-5228.

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Office of Nuclear Material Safety
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U. S. NUCLEAR REGULATORY COMMISSION

TECHNICAL EVALUATION REPORT FOR

THE U.S. DEPARTMENT OF ENERGY SAVANNAH RIVER SITE

DRAFT SECTION 3116 WASTE DETERMINATION FOR

SALT WASTE DISPOSAL

December 2005

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EXECUTIVE SUMMARY

In March 2005, the U.S. Department of Energy (DOE) submitted the "Draft Section 3116 Determination, Salt Waste Disposal, Savannah River Site" (SRS) for review by the U.S. Nuclear Regulatory Commission (NRC), as required by the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA). The draft waste determination addresses salt waste that DOE proposes to remove from the high-level waste (HLW) tank farms, treat through various processes, and dispose of on site in the Saltstone Disposal Facility (SDF). This Technical Evaluation Report (TER) presents information on DOE's salt waste processing strategy, the applicable review criteria, and the NRC's review approach, as well as the NRC's analysis and conclusions with respect to whether there is reasonable assurance that DOE's proposed approach can meet certain requirements of the NDAA for determining that waste is not HLW. The NRC is not providing regulatory approval of DOE's waste determination activities and DOE is responsible for determining whether the waste streams addressed in the draft waste determination are HLW.

Based on the information provided by DOE to the NRC in letters dated March 31, June 30, July 15, September 15, and September 30, 2005, the NRC staff has concluded that there is reasonable assurance that the applicable criteria of the NDAA can be met provided certain assumptions made in DOE's analyses are verified via monitoring. These assumptions, which are described in Section 4.3.1 of the TER, are important to demonstrating that the performance objectives in 10 CFR 61, Subpart C, can be met and fall into the following general categories: wasteform and vault degradation, the effectiveness of infiltration and erosion controls, and estimation of the radiological inventory. The NDAA requires NRC, in coordination with the State of South Carolina, to monitor disposal actions taken by DOE for the purpose of assessing compliance with the performance objectives of 10 CFR 61, Subpart C. Consequently, NRC requests that DOE develop proposed approaches that DOE will use to address the areas identified in Section 4.3.1. NRC will then, in coordination with the State, develop a program by which NRC will monitor DOE's implementation of the approaches.

There are 51 HLW tanks at the SRS site, two of which are operationally closed. The remaining 49 tanks currently hold approximately 1.34×10^{8} L (36.4 Mgal) of waste and 15.8×10^{18} Bq (426 MCi) of radioactivity. The tank waste consists of a mixture of insoluble metal hydroxide solids, referred to as sludge, and soluble salt supernate and crystallized salt, the combination of which is referred to as salt waste. The salt waste comprises approximately 1.25×10^{8} L (33.8 Mgal) (93% of the total in the tank farm) and 8.25×10^{18} Bq (223 MCi) (52% of the total in the tank farm). DOE estimates that approximately 1.1×10^{17} to $1.9 \text{ X } 10^{17}$ Bq (3 to 5 MCi) of salt waste would be disposed of in the SDF. The vast majority of the activity of the tank farm waste would be sent to the onsite Defense Waste Processing Facility (DWPF) for processing and eventual disposal in a Federal repository.

DOE intends to process the salt waste to segregate the low-activity fraction by using a two-phase, three-part approach. The first phase (called Interim Salt Processing) will involve two steps to treat the lower activity salt waste: (1) processing of a minimal amount of the lowest activity salt waste though a process involving Deliquification, Dissolution, and Adjustment (DDA) of the waste, and (2) processing of a minimal amount of additional waste with slightly higher activity levels using an Actinide Removal Process (ARP) and a Modular Caustic Side Solvent Extraction (CSSX) Unit (MCU), along with DDA. The second phase would involve the separation and processing of the majority of the salt waste using a much larger facility, the Salt Waste Processing Facility (SWPF).

The NDAA contains three criteria for determining that waste is not HLW. The first is that the waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW. This criterion allows for the consideration that waste may require disposal in a deep geologic repository even though the other criteria of the NDAA can be met. Consideration could be given to those circumstances under which geologic disposal is warranted to protect public health and safety and the environment; for example, unique radiological properties of the waste. Given that there is reasonable assurance that DOE's proposed approach can meet the other criteria in the NDAA, including the performance objectives of 10 CFR 61, Subpart C, and that there appears to be no other properties of the waste that would require deep geologic disposal, the NRC finds that there is reasonable assurance that Criterion One of the NDAA can be met.

The second criterion of the NDAA is that the waste has had highly radioactive radionuclides removed to the maximum extent practical. To assess conformance with Criterion Two, the NRC assessed the estimated waste inventory, the identification of highly radioactive radionuclides, radionuclide removal efficiencies, and the selection of treatment processes and approaches. NRC's conclusions regarding Criterion Two are based on the following assumptions: (1) the DDA and ARP/MCU interim treatment processes will not be used to treat significantly larger volumes or higher activity waste than proposed in the draft waste determination, (2) the actual radionuclide removal efficiencies of the ARP/MCU and SWPF treatment processes will meet or exceed the reported lower bounds of the projected removal efficiencies; (3) during the DDA process, dissolution of saltcake and solids settling will be performed such that the amount of sludge entrained in salt waste after DDA processing does not exceed DOE's estimate of 200 mg/L salt waste, which was identified by DOE as being a conservative estimate; (4) during DDA processing, the lower bound of 30% removal of supernate is met or exceeded for deliquification; (5) financial data used to support DOE's cost estimates are reasonable; (6) the proposed dates of operation of the ARP/MCU and SWPF facilities will be met; and (7) the Tank 48 waste can be processed safely in the Saltstone Processing Facility (SPF), and the wasteform will meet the Class C limits and perform as expected. Based on the assumptions listed above, the NRC has concluded that there is reasonable assurance that DOE can meet Criterion Two with its proposed approach.

The third criterion of the NDAA is that the waste is disposed of in compliance with the performance objectives contained in NRC regulations at 10 CFR 61, Subpart C. The NDAA provides for additional consultation with the NRC if the waste is found to exceed Class C concentration limits, which is not the case for the salt waste evaluated in the draft waste determination, and also provides for disposal pursuant to State-approved closure plans or State-issued permits. These types of State closure plans and permits are outside of NRC's authority and are not addressed in this TER. Subpart C of 10 CFR 61 sets requirements for protection of the public, the inadvertent intruder, and individuals during operation, and also provides for site stability. To assess conformance with Criterion Three, the NRC evaluated: (1) the waste classification, (2) DOE's performance assessment, (3) the source term, (4) infiltration and erosion control, (5) wasteform and concrete vault degradation, (6) release and near-field transport, (7) hydrology and far-field transport, (8) dose methodology, (9) the safety assessment and radiation protection program for individuals during operations, and (10) the strategy to ensure stability of the disposal site after closure.

NRC's conclusions regarding Criterion Three are based on the following assumptions: (1) more realistic modeling of waste oxidation and release of technetium from an oxidized layer of waste will result in predicted doses significantly lower than those projected in the DOE sensitivity analysis for 100% oxidation of the waste; (2) the hydraulic conductivity of degraded saltstone and vault concrete will not be larger than 1×10^{-7} cm/s (1×10^{-1} ft/yr); (3) field-scale physical properties (e.g., hydraulic conductivity, effective diffusivity) of as-emplaced saltstone are not significantly different from the results of laboratory tests of smaller-scale samples performed to date; (4) cracking from any mechanism will not be significantly more extensive than currently predicted by DOE; (5) the numerical modeling results for moisture flow through fractures in the concrete and saltstone located in the vadose zone will be confirmed by future model support (i.e., fracture flow will not occur); (6) the previous Documented Safety Analysis bounds the impacts to individuals during operations that may be estimated from the proposed waste streams; (7) active institutional controls will be maintained for 100 years; (8) current projections of the radiological concentrations of the waste are greater than or equal to actual concentrations for highly radioactive radionuclides; (9) future tests will confirm that the physical properties of samples that contain organic materials similar to Tank 48 waste are consistent with non-organic containing samples; (10) the erosion control design that DOE eventually implements will not deviate significantly from the information submitted to the NRC in response to the NRC's request for additional information; (11) development and use of accurate information for the moisture characteristic curves of concrete and saltstone will not significantly increase currently estimated release rates; and (12) gas phase transport of oxygen will not significantly increase oxidation of technetium in the saltstone, including transport in fractures.

Based on the assumptions listed above, the NRC has concluded that there is reasonable assurance that DOE can meet Criterion Three, which by reference incorporates the performance objectives of 10 CFR 61, Subpart C. DOE should assess the factors important to assessing compliance with 10 CFR 61, Subpart C, that the NRC staff has identified in this TER and carefully review the assumptions that the NRC conclusions are based upon. The DOE analysis demonstrates that more realistic modeling of infiltration, water contact with the waste, waste oxidation, and radionuclide release in an unsaturated and potentially fractured system is needed, because conservative modeling yields results that are higher than the dose limits provided in 10 CFR 61, Subpart C for protection of the public. Likewise, adequate model support is essential to ensuring that public health and safety can be protected with reasonable assurance. As discussed in detail in this TER, the deterministic DOE base case result does not have adequate model support and is not clearly conservative. Considering the uncertainty in many key parameters, DOE's current deterministic base case should not be used as the basis for developing inventory limits. A revised base case should be based on the projected average vault inventory and orientation of multiple disposal vaults, include the expected magnitude and timing of climate change from the natural cycling of climates, include the expected magnitude and rate of oxidation of waste, consider liquid and gas flow in fractures that may develop, and account for the questionable moisture characteristic curve information for concrete and saltstone that was used in the previous analysis.

For a broader and more detailed discussion of DOE's approach and NRC's analyses and conclusions, please see the appropriate sections of the TER, as the full discussion is not replicated here in the Executive Summary. All of the conclusions reached by the NRC are based on the draft Section 3116 waste determination dated March 2005, DOE's responses to NRC's request for additional information, supporting references, and information provided during meetings between DOE and NRC. If, in the future, DOE determines it is necessary to revise its assumptions, analysis, design, or waste management approach and those changes are important to meeting the criteria of the NDAA, DOE should consult once again with NRC regarding the findings in this TER. It should also be noted that NRC is providing consultation to DOE as required by the NDAA and the NRC is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW. This NRC assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental waste determinations at SRS or at other sites.

1. INTRODUCTION

1.1 Background

The Savannah River Site (SRS) is a 780-square kilometer (300-square mile) site located in western South Carolina and owned by the U.S. Department of Energy (DOE) (see Figure 1). The site began operation in 1951 and has produced nuclear material for national defense, research, medical, and space programs. The waste from reprocessing of spent nuclear fuel for defense purposes has been commingled with non-reprocessing waste resulting from the production of targets for nuclear weapons and production of material for space missions. Significant quantities of radioactive waste are currently stored on site in large underground waste storage tanks.

SRS has a total of 51 underground waste storage tanks, all of which were placed into operation between 1954 and 1986. Twenty seven of the tanks meet current U.S. Environmental Protection Agency (EPA) requirements for full secondary containment and leak detection. The remaining 24 tanks do not have full secondary containment and do not meet EPA requirements. Twelve tanks without secondary containment have a history of leakage, but sufficient waste has been removed from these tanks such that there are currently no active leak sites. Two of the tanks, which did not have secondary containment, have been closed and grouted. Of the 49 remaining operational tanks, 29 are in the H Tank Farm and 20 are in the F Tank Farm.

Approximately 1.34×10^{8} L (36.4 Mgal) of waste, containing 1.58×10^{19} Bq (426 MCi) of radioactivity, is stored in the tanks at SRS. The waste is a mixture of insoluble metal hydroxide solids, referred to as sludge, and soluble salt supernate. The supernate volume has been reduced by evaporation, which also concentrates the soluble salts to their solubility limits. The resultant solution crystallizes as salts, and the resulting solid is referred to as saltcake. The saltcake and supernate combined are referred to as salt waste, and comprise 1.25×10^{8} L (33.8 Mgal) (93% of the total) and contains 8.25×10^{18} Bq (223 MCi) (52% of the total). The sludge comprises 9.6×10^{6} L (2.6 Mgal) of waste (7% of the total) and contains 7.51×10^{18} Bq (203 MCi) (48% of the total).

DOE intends to remove, stabilize, and dispose of the waste, and close all 49 of the operational tanks. The sludge is currently being stabilized in the Defense Waste Processing Facility (DWPF) through a vitrification process that immobilizes the waste in a borosilicate glass matrix for eventual disposal in a Federal repository. In order to continue to have adequate tank farm space to support DWPF operations and start up of the Salt Waste Processing Facility (SWPF), DOE has indicated it needs to remove a portion of the salt waste in the near term. This salt waste will be treated and disposed of on site. DOE’s approach for treating and disposing of the salt waste is evaluated by the U.S. Nuclear Regulatory Commission (NRC) staff in this Technical Evaluation Report (TER).

1.2 DOE-SRS Salt Waste Processing Strategy

The 49 operational tanks at SRS are constructed of carbon steel and consist of four different types of designs. Type III, the newest design, has a full-height secondary tank and forced water cooling. There are 27 Type III tanks. Type I and II tanks have five-foot-high secondary

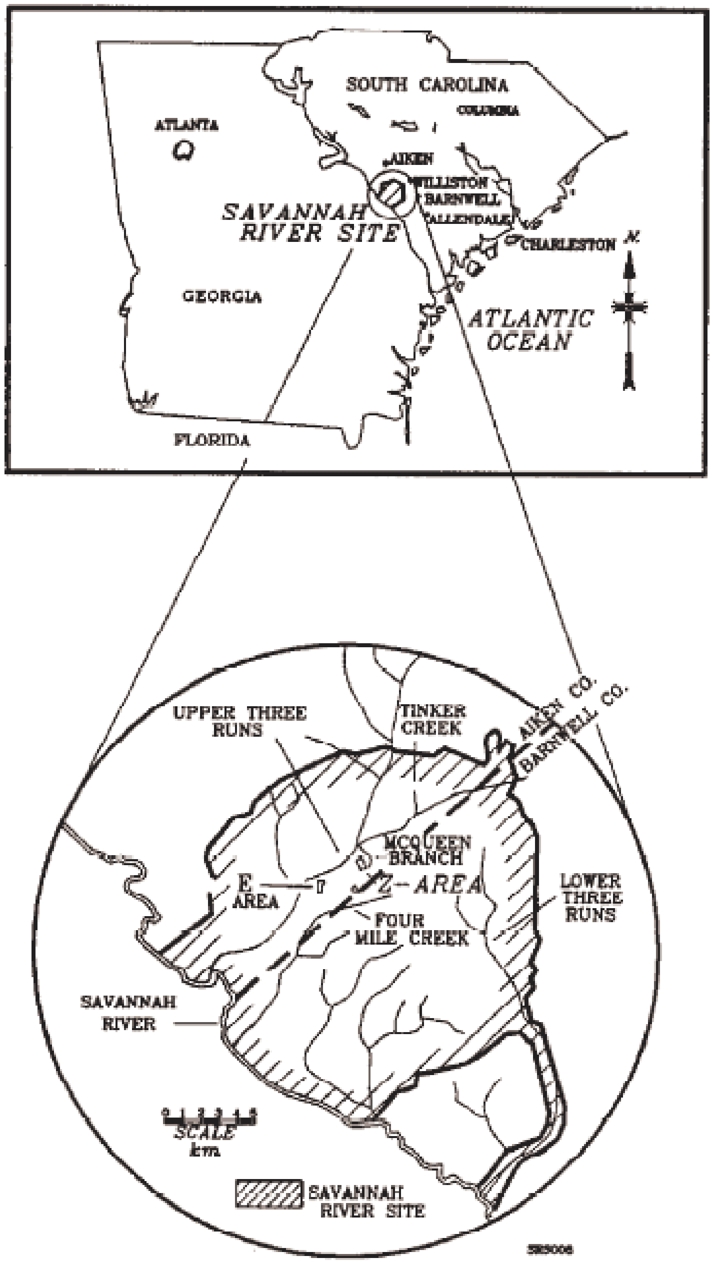

Figure 1 Location of the Savannah River Site

pans and forced cooling. Type IV tanks have a single steel wall and do not have forced cooling. Only Type III tanks meet EPA requirements for full secondary containment and leak detection. The waste in the 49 tanks consists of sludge, supernate, and saltcake. DOE intends to remove the salt waste and dispose of the low activity fraction on site in the Saltstone Disposal Facility (SDF).

DOE intends to separate the salt waste to segregate the low-activity fraction by using a two-phase, three-part approach (see Figure 2). The first phase (called Interim Salt Processing) will involve two parts to treat the lower activity salt waste: (1) processing of a minimal amount of the lowest activity salt waste through a process involving Deliquification, Dissolution, and Adjustment (DDA) of the waste, and (2) processing of a minimal amount of additional waste with slightly higher activity levels using an Actinide Removal Process (ARP) and a Modular Caustic Side Solvent Extraction (CSSX) Unit (MCU), along with DDA. The second phase would involve the separation and processing of the majority of the salt waste using the SWPF.

The DDA process will be the first interim process used and involves the following steps: (1) removing the supernate from above the saltcake; (2) extracting interstitial supernate from within the saltcake; (3) dissolving the saltcake and transferring the resulting salt solution to a settling tank; and (4) transferring the salt solution to the Saltstone Production Facility (SPF) for mixture with cement, slag, and fly ash to form a grout which is then disposed of in the SDF. DOE intends to use DDA on those tanks that contain the lowest activity material. In addition, during the same time period that DOE plans to use DDA, DOE plans to process a unique salt waste stream currently stored in Tank 48. This waste consists of approximately 8.8×10^5 L (0.24 Mgal) of salt solution containing 19,000 kg (8600 lbs) of potassium and cesium tetraphenylborate salts generated during an earlier, unsuccessful attempt to use a different technology for in-tank waste processing. DOE does not believe the waste in Tank 48 can be processed to remove radionuclides because of the explosion hazard posed by organic chemicals in the waste, and instead plans to combine the waste with another salt waste stream, currently planned to be the low-activity liquid recycle waste stream from DWPF, and transfer it to the SPF for processing and eventual disposal in the SDF. DOE indicated that the DDA process would be used from 2005 until approximately 2009, when the SWPF is expected to become operational.

The ARP/MCU process will be used to remove radionuclides in slightly higher activity salt waste than that proposed for the DDA process only. ARP/MCU will be used to treat some salt waste after it has undergone the DDA process. The ARP will use monosodium titanate (MST) strikes and cross-flow filtration to remove and concentrate the insoluble actinides. The high activity waste stream would be sent to the DWPF for vitrification and the low activity waste stream would be sent to the SPF for mixture with cement, slag, and fly ash and then disposal in the SDF. The MCU will use the CSSX process to remove cesium from the salt waste by using organic solvents to complex with cesium atoms in the waste stream. The solvent and salt solution will be fed to centrifugal contractors to ensure mixing, and the cesium will then be stripped and sent to DWPF for vitrification. The lower activity salt solution will be sent to the SPF for eventual disposal in the SDF. The ARP/MCU process would be used from 2007 until at least 2009, and may also be used to augment actinide removal capacity during operation of the SWPF.

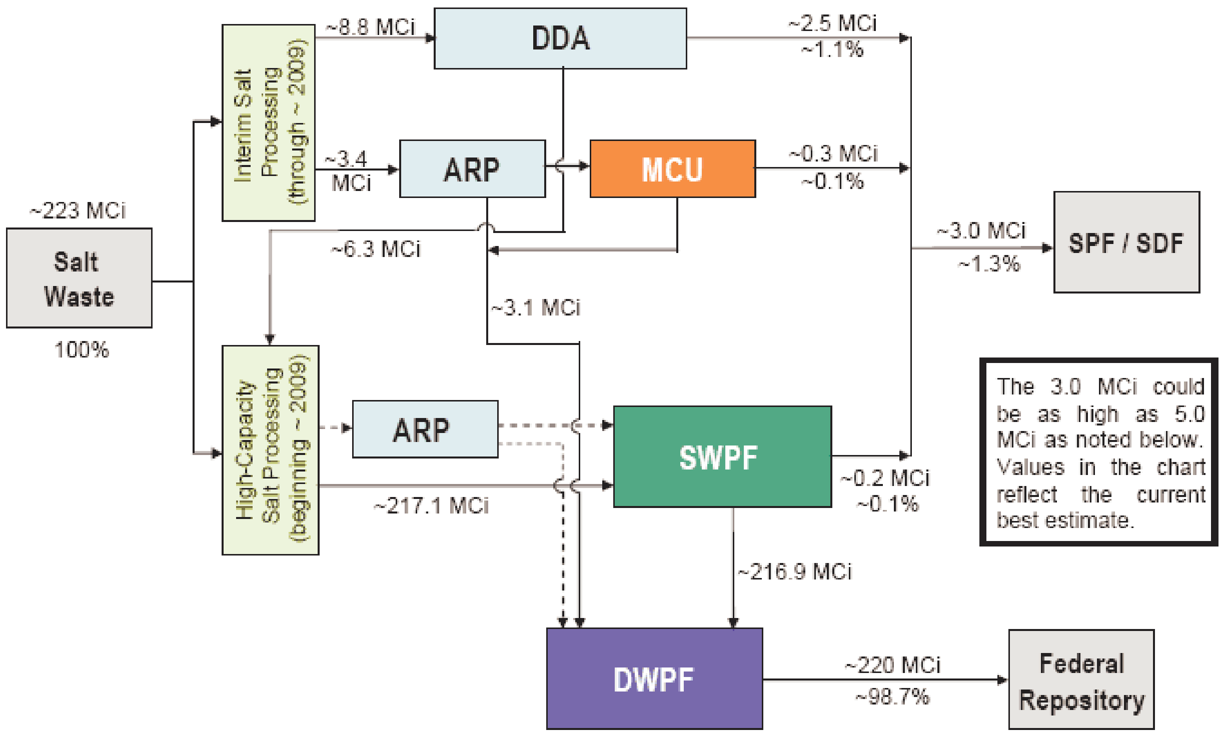

Figure 2 DOE Salt Waste Processing Strategy

The SWPF will use the same technologies as ARP and MCU to remove strontium, actinides, and cesium from the salt waste. However, due to its large scale, the SWPF will provide a much higher decontamination factor for cesium and have a higher throughput. The SWPF will consist of three operations: (1) Alpha Strike Process (ASP); (2) CSSX; and (3) Alpha Finishing Process (AFP). Like the ARP, the ASP will remove soluble strontium and actinides by sorption onto MST. The resultant slurry will be filtered to concentrate the MST and insoluble solids, with the filtrate sent forward through the SWPF and the concentrated MST/sludge sent to DWPF. The second SWPF operation will use the CSSX process to remove cesium, with the strip effluent sent to DWPF and the remaining waste stream sent to the SPF or to AFP if additional strontium and actinide removal is required to meet the SDF Waste Acceptance Criteria (WAC). AFP will be the final step in which an additional MST strike may be performed, if needed. The SWPF will be capable of processing 1.8×10^7 L (5 Mgal) of salt solution during the first year of operation and 2.2×10^7 to 3.3×10^7 L (6 to 9 Mgal) of salt solution per year each following year. The SWPF is planned to be operational by 2009 and continue to operate until the completion of salt processing in approximately 2019.

The SPF will mix the low-activity salt solution with dry chemicals (cement, slag, and fly ash) to form a grout mixture. The grout mixture will be pumped into engineered disposal vaults in the SDF, where the grout will solidify into a monolithic solid low-level waste form called “saltstone.” The SDF will consist of several large concrete vaults divided into cells. The saltstone grout itself provides primary containment of the waste, and the vaults provide secondary containment.

Several inches of uncontaminated grout will be poured on top of the saltstone grout to reduce skyshine. Vaults will be constructed as needed as salt waste processing continues.

Construction of the SPF and the first two vaults of SDF was completed between February 1986 and July 1998, and radioactive operations began in June 1990. Since that time, it has operated on an intermittent basis to treat and dispose of low-level waste from other on-site facilities. However, the SPF was originally designed to process wastes with lower activities than those currently proposed. Therefore, DOE is undertaking modifications to the design and operation of the facility, including additional shielding and equipment redesign to allow remote repair.

The SDF is located in the Z-area of SRS and is approximately 10 km (6.2 mi) from the nearest SRS site boundary on a well-drained local topographic high (see Figure 3). The Z-area is bounded by two streams: Upper Three Runs is approximately 1.6 km (1.0 mi) to the Northwest and McQueen Branch, a tributary to Upper Three Runs, is approximately 1.6 km (1.0 mi) to the East and 1.2 km (0.75 mi) to the Northeast. The facility has been sited and designed to ensure that even under extreme wet conditions, the water table will not rise to the base of the vaults. The Probable Maximum Flood on Upper Three Runs or its tributaries is 53 m (175 ft) above mean sea level which is below the planned maximum depth of the SDF vaults.

The two existing vaults (Vaults 1 and 4) are constructed of reinforced concrete containing slag. Vault 4 has dimensions of 60 m (197 ft) wide, by 180 m (590 ft) long, by 8 m (26 ft) high. The vault is divided into 12 cells of approximately 30 m (98 ft) by 30 m (98 ft). The vault is covered by a permanent roof with a minimum thickness of 10 cm (3.9 in) and a minimum slope of 0.19 cm/m (0.15 in/ft) from the midplane of the vault roof parallel to the short axis. The vault walls are approximately 0.46 m (1.5 ft) thick and the base mat is 0.61 m (2.0 ft) thick. Figure 4 provides a photograph of the existing SDF Vault 4. The SDF closure concept is described in Section 4.2.4. Up to 14 additional vaults may be required to dispose of the total volume of saltstone produced and the vaults will be covered by an engineered cap at final closure.

1.3 Waste Determination Criteria

Since 1969, the NRC has recognized the concept of incidental waste or waste incidental to reprocessing (WIR). The concept underlying WIR is that certain wastes can be managed based on their risk to human health and the environment, rather than the origin of the wastes. For wastes that originate from reprocessing of spent nuclear fuel, some are highly radioactive and need to be treated and disposed of as high-level waste (HLW). Other reprocessing waste does not pose the same risk to human health and the environment, and therefore does not need to be disposed of as HLW. This type of waste is termed “incidental waste” or “WIR.” DOE uses “waste determinations” to evaluate whether reprocessing waste is HLW or incidental waste.

The original incidental waste criteria were approved by the Commission in the Staff Requirements Memorandum (SRM) dated February 16, 1993, in response to SECY-92-391, “Denial of PRM 60-4 – Petition for Rulemaking from the States of Washington and Oregon Regarding Classification of Radioactive Waste at Hanford”. These criteria are described in the March 2, 1993, letter from R. Bernero, NRC, to J. Lytle, DOE as follows (NRC, 1993): (1) The waste has been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; (2) The waste will be

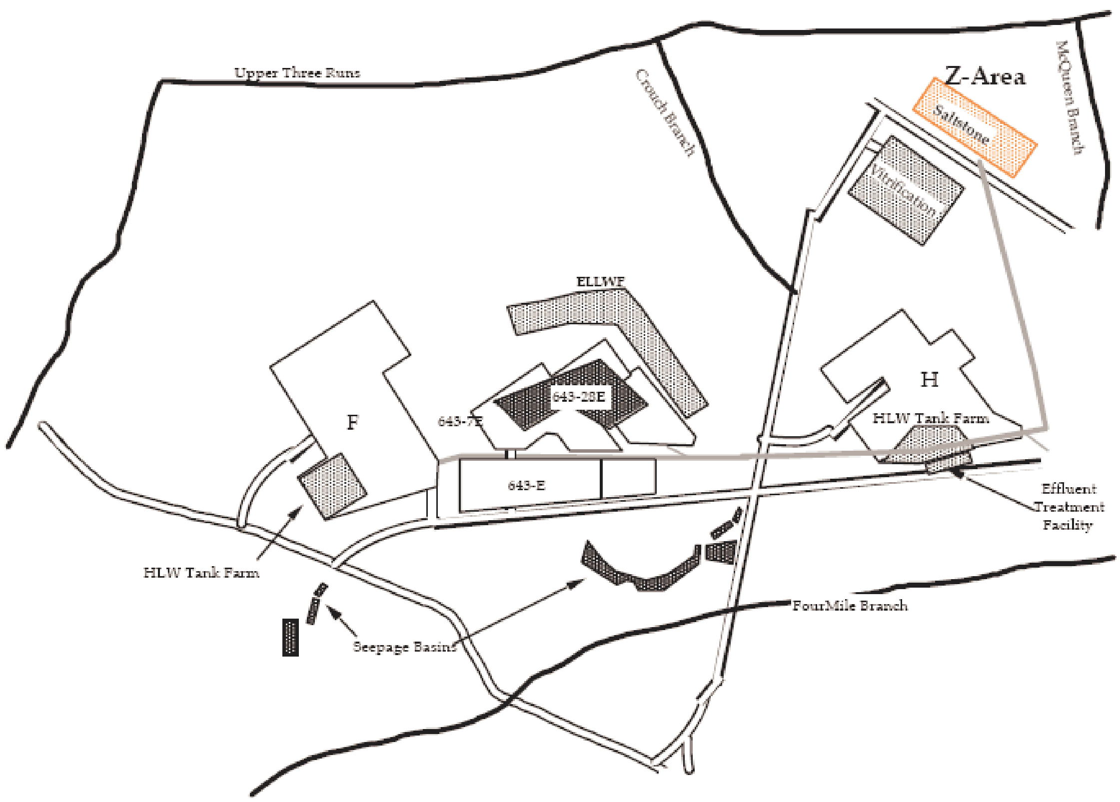

Figure 3 Location of the Z-Area

Figure 4 Photograph of Existing SDF Vault 4

incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR Part 61; and (3) The waste is to be managed, pursuant to the Atomic Energy Act, so that safety requirements comparable to the performance objectives set out in 10 CFR Part 61, are satisfied.

In the May 30, 2000, SRM on SECY-99-0284, “Classification of Savannah River Residual Tank Waste as Incidental,” the Commission indicated that a more generic, performance-based approach should be taken in regard to reviewing WIR determinations (NRC, 2000a). In effect, cleanup to the maximum extent that is technically and economically practical and demonstration that performance objectives could be met (consistent with those which the Commission demands for the disposal of LLW) should serve to provide adequate protection of the public health and safety and the environment. In the “Final Policy Statement for the Decommissioning Criteria for the West Valley Demonstration Project at the West Valley Site,” dated February 1, 2002, the Commission noted the criteria that should be applied to the incidental waste determinations at West Valley (NRC, 2002):

(1) The waste should be processed (or should be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; and

(2) The waste should be managed so that safety requirements comparable to the performance objectives in 10 CFR Part 61, Subpart C, are satisfied.

In October 2004, the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA) was signed into law. Section 3116 of the NDAA allows DOE to continue to use a process to determine that waste is not HLW and, among other things, requires that DOE consult with NRC on its non-HLW determinations. However, the NDAA is applicable only to South Carolina and Idaho and does not apply to waste transported out of those States. The NDAA establishes the following criteria for determining that waste is not HLW:

(1) The waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW;

(2) The waste has had highly radioactive radionuclides removed to the maximum extent practical; and

(3)(A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, and will 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; and
- (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or

(B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will 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;
- (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and
- (iii) pursuant to plans developed by the Secretary in consultation with the Commission

These are the criteria used by DOE in its draft waste determination for salt waste disposal.

1.4 NRC Review Approach

In February 2005, DOE-HQ and NRC established an Interagency Agreement (IA) that provided a mechanism for reimbursement for NRC activities in FY05, as required by the NDAA (NRC, 2005b). The IA also provides a statement of work that includes NRC's technical review of DOE's draft waste determination for salt waste processing at SRS.

NRC's review was based on DOE's "Draft Section 3116 Determination, Salt Waste Disposal, Savannah River Site." A non-publicly-available version of the draft waste determination was submitted by DOE on February 28, 2005, because DOE had not completed its required internal security review. On March 31, 2005, DOE submitted a publicly-available version of the draft waste determination, along with approximately 60 references, and it is that version on which the NRC staff based its review (DOE, 2005). The publicly-available version of the draft waste determination contained only minor, non-technical changes from the non-publicly-available version. The NRC staff performed a technical review of the information and sent a request for additional information (RAI) to DOE on May 26, 2005 (NRC, 2005a). The RAI contained questions concerning areas such as waste characterization, removal of highly radioactive radionuclides, engineered barriers, and saltstone degradation. NRC and DOE held open meetings on June 8 and June 20, 2005, to discuss DOE's preliminary proposed approaches to responding to the RAI. In letters dated June 30 and July 15, 2005, DOE submitted its RAI responses, as well as a Performance Objectives Demonstration Document (PODD), a revised Special Analysis to supplement the site's performance assessment (PA), and approximately 150 additional references. On July 27 and August 17-18, 2005, NRC and DOE met to discuss DOE's RAI responses. During those meetings, NRC requested that DOE provide additional information supporting the RAI responses. DOE responded by submitting "Response to Action Items from Public Meetings Between NRC and DOE to Discuss RAI for the Savannah River Site" in two separate letters dated September 15 and 30, 2005, along with approximately 50 supporting references. All of the documents submitted by DOE are publicly available and are referenced under the docket number PROJ0734 in the NRC's Agencywide Document Access and Management System (ADAMS).

NRC reviewed the draft waste determination and supporting documentation to assess whether it had sound technical assumptions, analysis, and conclusions with regard to meeting the criteria in the NDAA and thus, that DOE's proposed management of salt waste is protective of public health and safety and the environment. This approach is consistent with the one proposed by the NRC staff in SECY-05-0073, dated April 28, 2005, and approved by the Commission in the Staff Requirements Memorandum dated June 30, 2005. This TER addresses each of the applicable criteria in the NDAA and presents the NRC staff's approach, assumptions, and conclusions, as well as factors that the NRC staff believes may be important to meeting the performance objectives of 10 CFR 61, Subpart C.

NRC's conclusions are dependent on the assumptions discussed in the TER and if DOE revises its assumptions, analysis, design, or proposed waste management approach, DOE should consult once again with NRC regarding the findings in this TER. It should be noted that NRC staff is providing consultation to DOE as required by the NDAA and the NRC is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW. This NRC assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental waste determinations at SRS or at other sites.

1.5 Previous Reviews Performed for SRS

In December 1996, DOE requested that NRC review the DOE-SRS methodology for determining that residual tank waste was WIR. A Memorandum of Understanding between NRC and DOE was established in July 1997. Following the establishment of the MOU, NRC began its review of the "Regulatory Basis for Incidental Waste Classification at the Savannah River Site High-Level Waste Tank Farms." The NRC reviewed the information provided by DOE and issued a RAI in June 1998. In September 1998, DOE responded to the RAI and a public meeting was held in April 1999 to resolve some of the remaining outstanding issues. Following the public meeting, DOE submitted supplementary RAI responses in April 1999. NRC issued its final letter to DOE in June 2000. This previous WIR review used Criterion One and Criterion Three of the criteria established in 1993 (i.e., not the criteria in the NDAA) and was for waste remaining in tanks, not for waste being removed from the tanks and disposed of elsewhere. Therefore, this prior review is not directly applicable to the salt waste disposal draft waste determination.

2. CRITERION ONE

The waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW

2.1 Salt Waste Disposal

Criterion One allows for the consideration that waste may require disposal in a geologic repository even though the two other criteria of the NDAA may be met. Consideration could be given to those circumstances under which geologic disposal is warranted in order to protect public health and safety and the environment; for example, unique radiological characteristics of waste or non-proliferation concerns for particular types of material.

2.2 NRC Review and Conclusions (Criterion One)

Given the analysis in the following sections of this TER, which indicates that there is reasonable assurance that DOE can meet the applicable criteria in the NDAA based on certain assumptions, and the fact that there is no indication that other considerations would warrant disposal of the salt waste in a geologic repository, the NRC staff finds that there is reasonable assurance that Criterion One can be met by DOE.

3. CRITERION TWO

The waste has had highly radioactive radionuclides removed to the maximum extent practical

To evaluate compatibility with this criterion, NRC staff evaluated DOE's tank inventory estimates, estimated salt waste composition, process for identifying highly radioactive radionuclides, predicted radionuclide removal efficiencies, waste treatment technology selection, and alternatives to the proposed two-phase, three-part process.

3.1 Waste Inventory

The tank waste inventory (as of 12/1/04) was 1.34×10^{8} L (36.4 Mgal) containing 1.58×10^{19} Bq (426 MCi) of radioactivity. The radiological composition of the tank waste stored in the 49 operational underground carbon steel waste storage tanks can vary substantially from tank to tank and among the three primary physical phases of waste (salt supernate, saltcake, sludge). DOE’s practice at SRS has been to use evaporator systems to reduce the volume of supernate and concentrate the waste. During the evaporation process, salt waste forms two distinct phases: concentrated supernate solution and solid saltcake (collectively referred to as salt waste). Supernate is a concentrated salt solution (primarily nitrate and nitrite salts) that rests on top of the saltcake and fills the pore space of the saltcake. Saltcake is the crystallized precipitate formed when the salts reach their solubility limit. Because the concentrations of cesium and most other radioactive elements in the supernate are significantly lower than the concentrations of non-radioactive isotopes of sodium, they do not precipitate as salts and their concentrations in the saltcake are significantly lower than their concentrations in the supernate. Other than C-14, Na-22, and Al-26, radionuclides are not expected to be incorporated into salt crystals to a significant degree. Other radionuclides that are physically located in the saltcake layer are believed to be contained within entrained sludge. Sludge is a complex mixture of insoluble metal hydroxide solids that have precipitated in the tank. Although most of the sludge settles to the bottoms of the tanks, some insoluble metal hydroxides are suspended in the supernate and also incorporated in the saltcake.

To keep track of tank inventories, DOE has developed a Waste Characterization System (WCS), which is an electronic information system that is used to track waste tank data based on sample analyses, process histories, composition studies, and theoretical relationships. The system is used to track 96 chemicals and 41 radionuclides in each of the waste tanks. The system is updated routinely to reflect changes in tank inventory resulting from the receipt of new waste, tank to tank transfers, and evaporator operation. The WCS also is updated with the results of analyses of samples of supernate, salt, and sludge (Tran, 2005). In 2003 and 2004, a special campaign was completed to improve estimates of the radiological composition of the supernate and salt. The technical objectives of DOE's sampling approach are to understand the physical, chemical, and radiological characteristics of material stored in the waste tanks to support selection of appropriate options for waste treatment and disposal. In some cases sampling is limited because of access restrictions and costs associated with sampling waste from HLW tanks.

Sampling of the waste in the HLW tanks directly supports the projected composition of waste generated from the DDA, ARP/MCU, and SWPF for disposition in the saltstone facility. DOE estimates that the salt waste treatment and processing strategy will result in 6.27×10^8 L (170 Mgal) of grout containing 1.1×10^{17} Bq (3 MCi) of radioactivity to be disposed of in the SDF. The SDF has established WAC to ensure that the concentration of waste accepted by the facility is suitable for disposal. The saltstone facility WAC establishes the radionuclide limits for the waste, in addition to physical and chemical property limits. DOE plans to demonstrate compliance with the saltstone WAC prior to transferring waste from Tank 50 (the saltstone feed tank) to the SPF by sampling the waste in the feed tank or with material balance calculations and process knowledge (Ketusky, 2005). Compliance with the SDF WAC is discussed in further detail in Sections 4.2.2 and 4.2.3.

Table 1 provides the composition of supernate, sludge entrained in saltcake, and the sludge layer averaged over the two tank farms for the radionuclides considered in detail by DOE (see Section 3.3). However, waste composition varies significantly from tank to tank. For example, sludge formed from high-heat waste from the first canyon cycle typically has fission product concentrations that are approximately three orders of magnitude greater than the fission product concentrations in sludge formed from low-heat waste from the second canyon cycle (WSRC, 2005b).

In addition to the low-activity salt waste that will be processed during the interim period, DOE has proposed to create additional tank space by removing material from Tank 48, combining it with DWPF recycle, and directly sending the material to the SPF to be solidified into a low-activity saltstone wasteform with no additional processing. Tank 48 is a new style tank with a strategic location with respect to the salt processing facilities. It currently contains approximately 8.8×10^5 L (0.24 Mgal) of salt solution that contains approximately 19,000 kg (8600 lbs) of potassium and cesium tetraphenylborate (TPB) salts from the unsuccessful In-Tank Precipitation (ITP) process (Ledbetter, 2004; Fowler, 2004). The Tank 48 waste is estimated to contain 3×10^{16} Bq (0.8 MCi) of activity.

Table 1 Average Composition of Primary Physical Phases in DOE-SRS Tank Waste (Based on information in Tran [2005])

Radionuclide	Supernate (Ci/L)	Sludge Entrained in Saltcake (Ci/L)	Sludge Layer (Ci/L)
Cs-137*	2.43	8.19×10^{-3}	9.62×10^{-1}
Sr-90*	6.12×10^{-4}	1.88×10^{-1}	1.65×10^{1}
Se-79	9.84×10^{-7}	not reported	1.47×10^{-4}
Tc-99	3.65×10^{-4}	not reported	2.52×10^{-3}
I-129	1.97×10^{-7}	not reported	1.01×10^{-8}
Sn-126	4.97×10^{-6}	not reported	1.92×10^{-4}
U-232	3.17×10^{-10}	not reported	5.32×10^{-8}
U-233	2.93×10^{-8}	not reported	9.31×10^{-6}
U-234	4.64×10^{-8}	not reported	3.54×10^{-6}
U-235	9.24×10^{-10}	3.70×10^{-9}	1.51×10^{-7}
U-236	3.95×10^{-9}	not reported	5.82×10^{-7}
U-238	7.51×10^{-8}	8.30×10^{-8}	6.12×10^{-6}
Np-237	4.69×10^{-8}	not reported	9.36×10^{-6}
Pu-238	6.35×10^{-4}	1.20×10^{-3}	1.90×10^{-1}
Pu-239	3.73×10^{-5}	4.00×10^{-5}	4.53×10^{-3}
Pu-240	1.00×10^{-5}	not reported	1.96×10^{-3}
Pu-241	4.15×10^{-4}	not reported	1.18×10^{-1}
Pu-242	1.04×10^{-8}	not reported	3.10×10^{-6}
Am-241	4.01×10^{-6}	not reported	3.43×10^{-2}
Am-242m	2.31×10^{-9}	not reported	1.97×10^{-5}
Cm-244	1.60×10^{-6}	not reported	1.37×10^{-2}
Cm-245	1.58×10^{-10}	not reported	1.35×10^{-6}

* Includes daughter

3.2 NRC Evaluation – Waste Inventory

To estimate the inventory of various radionuclides in the tank farm system, DOE uses the WCS to account for additions of waste to the tank farm and transfers between tanks. The WCS system appears to be an effective tool to track the quantities and composition of the waste in the tank farm. However, a number of inputs to the WCS are based on assumptions, process knowledge, or other forms of indirect information. Despite campaigns to improve the estimates of radiological composition in the salt and supernate in 2003 and 2004, sampling remains limited. For example, although Cs-137 is relatively well characterized compared to other radionuclides, DOE estimates that, because of uncertainties in salt waste characterization, the total amount of activity sent to saltstone could be 70% higher than the expected value of 1.1×10^{17} Bq (3 MCi) (DOE, 2005). As discussed in Section 3.6, the NRC staff concludes that the greatest amount of uncertainty in the composition of the waste sent to the SDF results from uncertainties in the tank inventories and uncertainties in the results of the DDA process. NRC's evaluation of DOE's plan to control the uncertainty in the SDF inventory is discussed in Section 4.2.3.

3.3 Identification of Highly Radioactive Radionuclides

In its March 2005 submittal, DOE identified "radionuclides considered in detail" as Cs-137 (and its daughter Ba-137m), Sr-90 (and its daughter Y-90), Se-79, Tc-99, I-129, Sn-126, and actinides (isotopes of U, Pu, Am, Np, and Cm). DOE stated in its draft waste determination that the selection of radionuclides considered in detail was based on scientific expertise, knowledge, and health physics principles as applied to the SRS salt waste, and that the list of radionuclides considered in detail included those radionuclides in Tables 1 and 2 of 10 CFR 61.55 that are in the SRS salt waste in quantities such that they may be important to meeting the performance objectives in 10 CFR 61, Subpart C. Highly radioactive radionuclides were not explicitly defined in the March 2005 submittal (DOE, 2005). In its RAI, NRC asked DOE to provide a list of the radionuclides it classified as highly radioactive radionuclides in the context of the SRS saltstone waste determination (NRC, 2005a). In its RAI, NRC also indicated that the NRC staff believes that highly radioactive radionuclides are those radionuclides that contribute most significantly to risk to the public, workers, and the environment.

In response to NRC's RAI, DOE indicated that it believes highly radioactive radionuclides in the context of Section 3116 to be those radionuclides that, using a risk-informed approach, contribute most significantly to radiological risk to the workers, the public, and the environment (WSRC, 2005b). In its RAI response and supporting document, DOE indicated that it based its identification of highly radioactive radionuclides on the characteristics of saltstone formed from untreated salt waste (i.e., salt waste that had been solidified without application of DDA, ARP/MCU, or SWPF treatment). The approach developed by DOE involved two types of comparisons. First, DOE compared the concentrations of radionuclides in solidified untreated salt waste to NRC Class A concentration limits. Then, DOE calculated public, intruder, and site worker doses that would result if salt waste were solidified and placed in the SDF without treatment and compared each of these doses to 10% of the appropriate low-level waste performance objective (10 CFR 61.41, 61.42, or 61.43). Based on the first comparison, DOE concluded that Cs-137, Sr-90, and alpha-emitting transuranic nuclides were the only radionuclides in untreated solidified salt waste that would exceed NRC Class A concentration limits (10 CFR 61.55). Second, DOE used the pathway dose conversion factors it calculated for its deterministic base-case PA for Vault 4 to predict the dose to a member of the public from radionuclides in untreated solidified waste (Cook et al., 2005). DOE concluded that, in its base-case analysis, no radionuclides in untreated solidified salt waste would result in a dose to a member of the public that exceeded 10% of the 0.25 mSv/yr (25 mrem/yr) public dose limit (10 CFR 61.41). Third, DOE concluded that Cs-137 was the only radionuclide in untreated solidified waste that would exceed a dose of 0.50 mSv (50 mrem), or 10% of the 5 mSv (500 mrem) used in the Draft Environmental Impact Statement for 10 CFR 61 (see Section 4.2.18), to an inadvertent intruder. Fourth, DOE concluded that Cs-137 (including its daughter, Ba-137m) was the only radionuclide in untreated salt waste that would result in a worker gamma dose in excess of 500 mrem/yr, or 10% of the 50 mSv (5 rem/yr) limit (10 CFR 61.43). Finally, DOE noted that Sr-90, Cs-137, and alpha-emitting TRU nuclides were the only radionuclides driving worker inhalation dose. On the basis of these comparisons, in its RAI response (WSRC, 2005b), DOE identified Sr-90, Cs-137, and alpha-emitting transuranic nuclides (alpha-emitting isotopes of Pu, Am, Np, and Cm) as highly radioactive radionuclides in the context of the SRS saltstone Section 3116 draft waste determination.

In the RAI response (WSRC, 2005b) and supporting document (WSRC, 2005c), DOE indicated that, although Se-79, I-129, Tc-99, Sn-126, and uranium isotopes were radionuclides considered in detail in the original waste determination, they were not highly radioactive radionuclides in the context of the draft waste determination for SRS salt waste because DOE believed that concentrations of these nuclides in untreated salt waste were sufficiently low that they did not present a significant radiological risk to members of the public, site workers, or the environment. In addition, DOE noted that the concentrations of I-129 and Tc-99 in untreated salt waste are expected to be below NRC Class A limits (10 CFR 61.55). However, after the submittal of the RAI response, DOE added Se-79, I-129, and Tc-99 to its list of highly radioactive radionuclides for the draft waste determination because DOE determined that Se-79, I-129, and Tc-99 dominated the predicted doses to members of the public in several sensitivity analysis cases that DOE performed at the request of NRC (see Sections 4.2.15 and 4.2.16) (WSRC, 2005a). In addition, in its written response to action items from the public meetings between NRC and DOE to discuss the RAI (WSRC, 2005a), DOE indicated that only four alpha-emitting transuranic nuclides, Pu-238, Am-241, Cm-244, and Pu-239, were being retained on its list of highly radioactive radionuclides in the context of the Section 3116 draft waste determination for SRS salt waste. In a supporting document (WSRC, 2005c), DOE indicated that these four nuclides account for approximately 99% of the transuranic alpha activity in untreated salt waste. Thus, DOE's final list of highly radioactive radionuclides in the context of the Section 3116 draft waste determination for SRS salt waste includes Cs-137 (and its daughter, Ba-137m), Sr-90 (and its daughter, Y-90), Pu-238, Am-241, Cm-244, Pu-239, Se-79, I-129, and Tc-99.

3.4 NRC Evaluation – Identification of Highly Radioactive Radionuclides

The definitions of "highly radioactive radionuclides" used by NRC staff and by DOE appear to be consistent. Specifically, NRC staff and DOE agree that highly radioactive radionuclides are those radionuclides that contribute most significantly to radiological risk to the public, workers, and the environment. To identify highly radioactive radionuclides, DOE compared the inventories of radionuclides in untreated salt waste (i.e., salt waste solidified without application of DDA, ARP/MCU, or SWPF treatment) to NRC Class A concentration limits (WSRC, 2005c). DOE also compared predicted doses resulting from untreated solidified salt waste to workers, members of the public, and inadvertent intruders to NRC limits (WSRC, 2005c). Based on this approach, DOE identified highly radioactive radionuclides as Cs-137 (and its daughter, Ba-137m), Sr-90 (and its daughter, Y-90), and alpha-emitting isotopes of Pu, Am, Np, and Cm. Based on the results of sensitivity analyses performed after the RAI response submittal, DOE added Se-79, Tc-99, and I-129 to its list of highly radioactive radionuclides (WSRC, 2005a). NRC staff evaluated DOE's final selection of highly radioactive radionuclides and has concluded that the combination of approaches used by DOE to identify highly radioactive radionuclides in the context of the draft waste determination for SRS saltstone waste is appropriate.

Although DOE identified Se-79, Tc-99 and I-129 as highly radioactive radionuclides, DOE also concluded that “the associated risks of Se-79, Tc-99 and I-129 are so low that it would not be sensible or reasonable to target these radionuclides for further removal” (WSRC, 2005a). While the NRC staff agrees that DOE’s base-case PA results are consistent with this conclusion (see Sections 4.2.15 and 4.2.16), the NRC staff believe that the consideration given to sensitivity and uncertainty analysis results must be related to the amount of model support provided for the base-case results. As discussed in Section 4.2.16, the NRC staff has concluded that there are realistically conservative cases in which doses due to Se-79 and I-129 may be significantly greater than DOE’s predicted base-case result. In addition, as discussed in Section 4.2.16, the NRC staff has concluded that there may be more uncertainty in the potential dose from Tc-99 than appears to be recognized by DOE. The NRC staff concludes that DOE’s use of sensitivity analysis to identify highly radioactive radionuclides is appropriate and that it is inappropriate to discount potential doses from highly radioactive radionuclides solely on the basis of base-case results, especially in cases in which the base-case has insufficient support (see Section 4).

3.5 Radionuclide Removal Efficiencies

Section 1.2 provides an overview of DOE’s strategy for salt waste processing. As previously described, DOE intends to process the salt waste to segregate the low-activity fraction by using a two-phase, three-part approach (see Figure 2). The first phase (called Interim Salt Processing) will involve two parts to treat the lower activity salt waste: (1) processing of a minimal amount of the lowest activity salt waste through the DDA process, and (2) processing of a minimal amount of additional waste with slightly higher activity levels using the ARP and MCU, along with DDA. The second phase would involve processing of the majority of the salt waste using the SWPF.

Each part of the salt waste processing strategy (DDA, ARP/MCU, SWPF) involves unit operations and processes capable of removing highly radioactive radionuclides. Table 2 provides an overview of the activities of waste DOE plans to treat with the proposed salt waste treatment processes. The activity totals are dominated by the fission products Cs-137 and Sr-90 and their daughters. Because treatment efficiency in each of the primary processes is sensitive to the physical phase of the radionuclides in the waste and because some of the unit processes are radionuclide-specific, the removal efficiencies for individual radionuclides considered in detail may be different than the total activity removal efficiencies shown in Table 2. Table 3 provides the expected efficiencies for removal of radionuclides DOE considered in detail in the draft waste determination by each of the primary treatment processes. In general, more soluble radionuclides (e.g., I-129, Tc-99, Cs-137) can only be removed by chemical processes whereas relatively insoluble radionuclides (e.g., Sr-90,

Table 2 Overview of Salt Waste Processing Efficiencies

Process	Total Input (MCi)	Total Output to SDF (MCi)	Removal Efficiency (%)
DDA	8.8	2.5	79 [(3)]
ARP/MCU	3.4	0.3	88
SWPF	217.1	0.2	99.9
Total	223 [(1)]	3.0 [(2)]	98.7

(1) The total is not the sum of the column values because some output from DDA will be sent to the SWPF
(2) Because of uncertainty in the inventory, this value may be as high as 5.0 MCi
(3) As discussed in the text, the removal efficiency reported by DOE has been adjusted so that it does not reflect the contribution of Tank 48 waste.

actinides) also can be removed by physical processes such as settling and filtration. Some radionuclides considered in detail (e.g., Se-79, Sn-126) have significant fractions in both the liquid and solid phases.

Of the 9.3 x 10^{16} Bq (2.5 MCi) that DOE predicted will be sent to SPF from the DDA process, 3 x 10^{16} Bq (0.8 MCi) originates in Tank 48 waste (DOE, 2005). The waste in Tank 48 is incompatible with the existing treatment processes because of organic components of the TPB salts. Flammable vapors (benzene) are produced from the decomposition of TPB, and the rate of decomposition increases with increasing temperature and the presence of catalysts. The presence of catalysts in other tank farm wastes and the need to maintain relatively low temperatures precludes mixing the Tank 48 waste with other waste or processing it through the proposed treatment systems. Because of the potentially problematic interactions of Tank 48 waste with other tank farm waste or with the proposed treatment processes, Tank 48 waste will be sent to SPF without treatment; that is, there will be no removal of highly radioactive radionuclides from the Tank 48 waste before it is solidified. In Table 2, the removal efficiency reported by DOE for the DDA process, which included Tank 48 waste as part of the DDA total, has been adjusted to reflect activity removal only in the waste that will be treated with the DDA process.

Table 3 shows predicted efficiencies of removal of radionuclides considered in detail by DOE. DOE's predicted removal efficiencies are shown for the radionuclides for which removal efficiencies were provided. For radionuclides for which DOE did not supply predicted removal efficiencies, NRC staff estimated removal efficiencies. The methods and assumptions that NRC staff used to calculate removal efficiencies are discussed in Section 3.6. DOE predicts that two-thirds of the entrained sludge in salt waste treated with the DDA process will be removed by settling after saltcake dissolution. In addition, DOE predicts that the deliquification process will remove half of the interstitial supernate in saltcake (Pike, 2005). In calculating DDA removal efficiencies for Cs, Sr, and actinides, DOE estimated that 66% of the insoluble radionuclides and 50% of the dissolved radionuclides would be removed by the DDA process (WSRC, 2005c). DOE predicted that approximately 100% of the insoluble radionuclides would be removed by filtration during the ARP/MCU and SWPF processes. In addition, DOE predicted that 91% of the dissolved Cs, 99.2% of the dissolved Sr, 92.3% of the dissolved Pu, 41% of the dissolved Am, and 41% of the dissolved Cm would be removed during ARP/MCU

Table 3 Projected Removal Efficiencies of Radionuclides Considered in Detail (in %)

Radionuclide	DDA [1,2]	ARP/MCU [2,4,5]	SWPF [5]	Overall [7]
Cs-137	50 [3]	91 [3]	99.998 [3]	99
Sr-90	66 [3]	99.997 [3]	99.98 [3]	> 99
Se-79	40	59	59	60 [8]
Tc-99	4.2	6.3	6.3	5.7
I-129	0.03	0.05	0.05	< 0.05 [9]
Sn-126	18	27	27	27
U-232	46	[4]	72	72
U-233	53	[4]	82	80
U-234	36	[4]	58	47
U-235	46	[4]	71	70
U-236	45	[4]	70	66
U-238	37	[4]	59	58
Np-237	54	[4]	86	84
Pu-238	63 [3]	98.1 [3]	95.5 [3]	94
Pu-239	59 [3]	96.4 [3]	91.6 [3]	91
Pu-240	58	97	94	93
Pu-241	60	98	95	95
Pu-242	60	98	96	95
Am-241	66 [3]	99.3 [3]	99.7 [3]	> 99
Am-242m	66	99	99.8	> 99
Cm-242	66	99	99	> 99
Cm-243	66	99	99	> 99
Cm-244	66 [3]	99.3 [3]	99.8 [3]	99
Cm-245	66	98	99	99
alpha-emitting TRU	63 [3]	98.1 [3]	96 [3]	95

(1) Unless otherwise noted, removal efficiencies were calculated by assuming removal of 66% of the insoluble fraction of radionuclides and SWPF removal efficiencies of 50% of the soluble radionuclides, as discussed in Section 3.6.
(2) Unless otherwise noted, removal efficiencies were calculated based on the phase partitioning of average salt waste (WSRC, 2005c). As discussed in Section 3.6, actual removal efficiencies of the DDA and ARP/MCU processes may differ significantly from the estimated removal efficiencies for radionuclides for which phase partitioning in the average untreated salt waste is not representative of phase partitioning in the waste sent to DDA or ARP/MCU.
(3) Removal efficiency reported by DOE (WSRC, 2005c).
(4) Insufficient information was provided for NRC staff to estimate a removal efficiency, as discussed in Section 3.6.
(5) Unless otherwise noted, removal efficiencies were calculated based on removal of 100% of the insoluble fraction and DOE's predicted removal efficiencies for soluble radionuclides for the appropriate process (WSRC, 20005c).
(6) As described in Section 3.6, the removal efficiencies for dissolved radionuclides were those for the use of ARP with MST (WSRC, 2005c).
(7) Unless otherwise specified, overall removal efficiencies were calculated based on the inventory of radionuclides in salt waste (WSRC, 2005c) divided by the projected inventory in the SDF (d'Entremont and Drumm, 2005).
(8) The overall removal efficiency appears to be greater than removal efficiency of individual processes because of differences in rounding.
(9) Removal efficiency was calculated based on removal of insoluble I-129 because the removal efficiency calculated based on the predicted saltstone inventory was affected by a rounding error.

treatment (WSRC, 2005c). DOE predicted that 99.998% of the dissolved Cs, 95% of the dissolved Sr, 82% of the dissolved Pu, 78% of the dissolved Am, 58% of the dissolved Np, and 26% of the dissolved U would be removed during SWPF treatment (DOE, 2005). DOE indicated that dissolved Se, I, Tc, and Sn were not specifically targeted for removal by salt waste treatment processes (DOE, 2005).

The ARP/MCU process can be used with or without MST strikes to remove soluble Sr and actinides. The removal efficiencies used to calculate the removal of dissolved radionuclides were those predicted for the use of ARP with MST (WSRC, 2005c) because DOE has stated that it intends to use MST strikes for all batches of waste processed with ARP as long as the necessary processing throughput can be maintained (WSRC, 2005b). However, if the target throughput cannot be maintained and if the solidified waste would still meet Class C concentration limits, ARP may be run in a “filtration only” mode without MST strikes. The primary effect of not using MST strikes during ARP treatment would be to lower the expected efficiencies of removal of Pu-238 (from 98.1% to 75%), Pu-239 (from 96.4% to 54%), and total alpha-emitting TRU (from 98.1% to 78%). The removal efficiencies of Sr-90, Am-241, and Cm-244 would decrease slightly but would remain above 98%.

3.6 NRC Evaluation – Radionuclide Removal Efficiencies

Table 3 shows predicted efficiencies of removal of radionuclides considered in detail by DOE. As discussed in Section 3.5, DOE estimated that 66% of the insoluble radionuclides and 50% of the dissolved radionuclides would be removed by the DDA process (WSRC, 2005c). This approach to the calculation of the removal of dissolved radionuclides by the DDA process appears to underestimate DDA removal of Cs-137 and overestimate DDA removal of other dissolved radionuclides. The approach appears to underestimate the removal of Cs-137 because the factor of 0.5 is applied to the entire dissolved fraction of Cs-137 even though the free supernate is expected to be removed entirely rather than only being approximately 50% removed as the interstitial supernate is (Pike, 2005). For example, the May 2005 WCS report indicates that approximately 30% of the total supernate in the tanks DOE plans to treat with DDA (Tanks 25, 28, and 41) is free, rather than interstitial (Tran, 2005). If it is assumed that 30% of the supernate is free and that 100% of the free supernate is removed and later treated at the SWPF, the nominal efficiency of removal of Cs-137 by DDA would be 65% rather than DOE's reported value of 50%. DOE's approach is also based on the assumption that all of the dissolved radionuclides in the interstitial supernate removed from the saltcake during deliquification are subsequently removed from the waste stream and are not sent to the SDF. This assumption is appropriate for Cs because the supernate removed from the tanks by deliquification is held for treatment in the SWPF, which, DOE predicts, will remove 99.998% of Cs-137. However, this approximation would overestimate the removal of other soluble radionuclides that are not removed by downstream processes. For example, I-129 is present almost entirely in supernate, but is not removed during SWPF treatment; therefore, although approximately half of the I-129 may be removed from the saltcake during deliquification, it is not removed from the extracted supernate during subsequent processing and ultimately is sent to the SDF.

In calculating removal efficiencies, DOE also made the approximation that the phase partitioning of radionuclides in waste being sent to each primary process was well represented by the phase partitioning of the radionuclides in the average untreated salt waste (WSRC, 2005c). While this is a reasonable approximation for the calculation of SWPF removal efficiencies because most of the waste will be processed through SWPF, it is a more tenuous assumption for both DDA and ARP/MCU processes because a relatively small fraction of the waste will be processed through these processes. If the partitioning of a radionuclide between the solid and liquid phases in batches of waste treated with the DDA and ARP/MCU processes is significantly different than the partitioning of the radionuclide in the average untreated salt waste, the actual removal efficiencies may be significantly different than the projected removal efficiencies shown in Table 3. This potential discrepancy is unlikely to affect the observed removal efficiencies of radionuclides that are either strongly soluble (e.g., Cs-137, I-129, Tc-99) or strongly insoluble (e.g., Sr-90, actinides), but could significantly affect the observed removal efficiencies of radionuclides whose partitioning can vary significantly from tank to tank (e.g., Se-79).

For radionuclides for which DOE did not provide DDA removal efficiencies, NRC staff calculated removal efficiencies based on the following assumptions: (1) 66% of the insoluble fraction of each radionuclide is removed through settling, (2) half of the total volume of free and interstitial supernate is removed by deliquification, as in DOE's calculation, and (3) the predicted efficiencies of the removal of dissolved radionuclides from supernate separated from saltcake during the DDA process are given by the removal efficiencies DOE reported for the removal of dissolved radionuclides by the SWPF. Neither of the assumptions about the removal of dissolved radionuclides has a significant effect on the predicted removal of Sr-90 and the actinides by the DDA process because they are relatively insoluble and are removed primarily through settling. Furthermore, assumptions about the amount of supernate that is removed during the DDA process do not affect the predicted removal of highly soluble radionuclides that are not removed by SWPF (e.g., I-129 and Tc-99). Because of these insensitivities to the assumptions discussed, for radionuclides other than Cs-137, calculated removal efficiencies were within 4% of DOE estimates irrespective of whether it was assumed that all of the supernate was interstitial or that 30% was free.

NRC staff used a similar method to calculate removal efficiencies for the ARP/MCU and SWPF processes. For isotopes for which removal efficiencies were not provided by DOE, NRC staff calculated removal efficiencies based on the removal of 100% of the insoluble fraction of the isotope and the ARP/MCU or SWPF removal efficiency that DOE predicted for the soluble phase of the appropriate element (DOE, 2005; WSRC, 2005c). Because DOE did not provide efficiencies of removal of dissolved U or Np by the ARP/MCU process, NRC staff could not calculate predicted removal efficiencies of isotopes of these elements by the ARP/MCU process. Because DOE indicated that dissolved Se-79, I-129, Tc-99, and Sn-126 were not targeted for removal by salt waste treatment processes, and that removal of these radionuclides would occur primarily because of removal of insoluble radionuclides (DOE, 2005), NRC staff assumed that neither the ARP/MCU nor the SWPF processes would remove any of the dissolved fraction of these radionuclides.

For comparison, NRC staff calculated an "overall" removal efficiency for each radionuclide (Table 3) based on the predicted inventory of a radionuclide in salt waste (WSRC, 2005c) divided by the projected inventory of the radionuclide in the SDF (d'Entremont and Drumm, 2005). Although DDA treatment is expected to remove a lower fraction of most radionuclides than the ARP/MCU and SWPF processes, the overall treatment efficiency for many radionuclides is similar to the SWPF removal efficiency because a relatively small fraction of the total inventory of each radionuclide is expected to be processed with DDA alone (Table 2).

In addition to uncertainties in tank inventories, uncertainties in SDF inventory will reflect uncertainties in the performance of each waste treatment process. Uncertainties in the extent of deliquification that will be achieved, the amount of sludge that will become entrained in salt waste, and the effectiveness of the DDA settling step to remove entrained sludge from DDA waste will contribute to the uncertainty in concentrations of radionuclides in DDA waste sent to the SDF. DOE estimated that deliquification will remove from 30% to 70% of the interstitial supernate and that settling will remove from 50% to 85% of the entrained sludge from the salt waste and concluded that the overall removal efficiency of the DDA process varied by approximately ±20% of the nominal value for both soluble and insoluble radionuclides (WSRC, 2005c). DOE also noted that the uncertainty in the amount of sludge that would be entrained in salt waste is more difficult to estimate, and that the concentration of entrained sludge in DDA waste prior to settling could vary from approximately 0 milligrams of sludge per liter salt waste to several grams of entrained sludge per liter salt waste (WSRC, 2005b). The uncertainty associated with the extent of radionuclide removal by the ARP/MCU is on the order of only a few percent for Cs-137, Sr-90, Am-241, and Cm-241, but is greater for Pu-238 and Pu-239 because Pu is more evenly partitioned between the sludge and supernate; the uncertainty in the predicted efficiency of removal of both Pu-238 and Pu-239 by the ARP/MCU process is approximately 25% if MST sorption is not used and approximately 10% if MST sorption is used. Reported uncertainties in the efficiency of removal of radionucides by the SWPF are less than 1% for Sr-90 and Cs-137, and approximately 5% or less for Pu-238, Pu-239, Am-241, and Cm-244.

Based on the information about uncertainties in tank inventories and removal processes provided by DOE, the NRC staff concludes that the greatest amount of uncertainty in the composition of the waste sent to the SDF results from uncertainties in the tank inventories and uncertainties in the results of the DDA process. NRC staff note that uncertainties in the DDA process are expected to have the largest effect on the SDF inventories of radionuclides that the ARP/MCU or SWPF treatment processes remove with a high removal efficiency. For example, because DOE predicts that approximately 90% of the Cs-137 in the SDF will originate from the DDA process (d'Entremont and Drumm, 2005), and that Cs-137 is primarily partitioned into the supernate (WSRC, 2005c), the amount of deliquification achieved during the DDA process will have a significant effect on the SDF inventory of Cs-137. Similarly, DOE predicts that approximately 77% of the Sr-90 in the SDF will originate from the DDA process (d'Entremont and Drumm, 2005). Because DOE predicts that approximately 100% of the Sr-90 in salt waste will be insoluble (WSRC, 2005c), the SDF inventory of Sr-90 will be sensitive to the amount of sludge entrained during the DDA process and the effectiveness of settling during the DDA process. Thus NRC staff concludes that efforts to maximize the amount of deliquification achieved, to minimize the amount of sludge entrained during DDA, and to maximize the amount of sludge that settles during DDA processing will have a significant effect on the inventory of highly radioactive radionuclides in the SDF and the demonstration that highly radioactive radionuclides have been removed to the maximum extent practical. Uncertainties in the SDF source term and process for determining whether waste will meet the SPF WAC are discussed with respect to the source term used in the PA calculations (Sections 4.2.2 and 4.2.3).

3.7 Selection of Treatment Processes

In developing its strategy for salt waste processing and disposal, DOE evaluated technologies available to remove radionuclides from salt waste. In addition, DOE compared alternatives to the proposed two-phase, three-part approach by evaluating cases in which waste processing would be delayed until either the ARP/MCU process or the SWPF becomes operational.

3.7.1 Alternative Treatment Technologies

The CSSX technology used in both the SWPF and MCU process for removal of Cs-137 was selected from approximately 140 alternatives that fell into 11 general technological categories (WSRC, 1998a). Based on the results of the original evaluation, conducted in 1998, DOE initially concluded that the CSSX alternative was technologically immature (DOE, 2001a). However, DOE resumed consideration of CSSX based on subsequent research and development results and a recommendation by the National Academy of Sciences (NAS) (NAS, 2000). In the original 1998 evaluation, technology selection was based on expert judgment. Each technology was judged based on the unknowns associated with design and implementation (i.e., level of development of the technology), risk of not meeting the waste treatment goals, potential risks to meeting the necessary budget and schedules, and safety risks, including nuclear safety risks, process hazards, and potential accidents (WSRC, 1998a). In 2000 and 2001, the NAS reviewed DOE's technology evaluations and selections (NAS, 2000; NAS, 2001). NAS concluded that a reasonable set of Cs removal technologies had been evaluated and provided input for continued technology development.

Both the ARP/MCU and SWPF removal process will remove Sr-90 and actinides by sorption onto MST and subsequent filtration. The use of MST sorption for Sr-90 and actinide removal was part of the ITP process that DOE originally planned to use to treat salt waste. Each of the four alternative Cs removal technologies discussed in the 2001 Savannah River Site Salt Processing Alternatives Final Supplemental Environmental Impact Statement (DOE, 2001b) was paired with MST sorption for Sr and actinide removal.

As discussed in Section 2.1, the DDA process relies on deliquification to remove Cs-137, which is primarily dissolved in supernate, from the saltcake for subsequent treatment by the SWPF. DOE considered using other technologies for deliquification, including using positive pressure and flushing. However, no alternate technologies were determined to be viable because of safety concerns, tank space limits, or chemistry constraints. In addition, the DDA process relies on settling to remove entrained sludge from the dissolved saltcake. In 2003, DOE evaluated both settling and various filtration options for the removal of insoluble radionuclides present in entrained sludge. Although DOE expects filtering to achieve approximately 100% removal of insoluble radionuclides and expects settling to remove only approximately 66% of insoluble radionuclides, DOE chose to use settling during DDA because DOE determined that filtration facilities could not be constructed and become operational by the time DDA processing was required (WSRC, 2005b).

In its response to the action items, DOE indicated that it did not target Se-79, Tc-99, or I-129 for removal but that the solid portion (approximately 60% of Se-79, 6% of Tc-99, and 0.05% of I-129) would be removed by a combination of settling and cross-flow filtration (see Section 3.5). DOE stated that removal of the dissolved portion of these radionuclides is impractical because of the low technical maturity of applicable technologies. DOE also stated that they have not conducted significant research and development activities regarding removal of these radionuclides from tank waste (WSRC, 2005a).

In addition to attempting to optimize technology selection, DOE plans to limit the radiological contribution from both the ARP/MCU and DDA interim processes by using these processes to treat only the lowest activity tank waste. For example, DOE has indicated that, within operational constraints, tanks with the lowest activity supernate will be selected for DDA processing. In addition, DOE indicated that tanks that have received high-heat waste will not be selected for DDA processing because sludge formed from high-heat waste has fission product concentrations that are three orders of magnitude greater than fission product concentrations in sludge formed from low-heat waste (WSRC, 2005b).

DOE evaluated several alternative technologies for processing Tank 48 waste. Options for in-situ organic destruction were considered, including hydrolysis, both chemical and thermal, and catalysis (e.g., palladium). However, the Tank 48 waste is contained in a carbon steel tank and therefore cannot be acidified, as each of these options would require, because of the potential for tank corrosion. DOE also considered treating the waste outside of the tanks using existing facilities. The waste cannot be processed in the DWPF melter because the breakdown of TPB in sufficient quantities could pose safety concerns. Construction of a separate waste treatment unit to remove the organics was considered but was found to be extremely costly and would require years to build and permit. DOE concluded that there are currently no proven, practical options for removing additional radionuclides from the Tank 48 waste or treating the organics to allow processing through the DDA, ARP/MCU, or SWPF. In the review of information submitted in response to the RAI, NRC staff identified a report that indicated that in-tank peroxide oxidation of Tank 48 waste may be technically and economically practical (Lambert et al., 2005). However, a separate DOE report stated that initial corrosion testing performed to support the technical evaluation of the process indicated that below a pH of 11, pitting corrosion of the tank may occur. Therefore the option was not technically practical (Zapp, 2004).

3.7.2 Alternatives to the Proposed Two-Phase Approach

DOE proposes to implement the selected technologies in a two-phase, three-part approach. In general, this approach is consistent with recommendations of the Defense Nuclear Facilities Safety Board (DNFSB) (DNFSB, 2001) and NAS (NAS, 2000), which both suggested that DOE should evaluate direct disposal of low activity wastes as an alternative to treating all of the waste with the same processes. Initially, DOE's primary consideration for removal of highly radioactive radionuclides was that substantial amounts of tank space are required to remove and prepare higher activity tank waste for disposal. Creation of sufficient tank space to allow DOE to maintain sustained risk-reduction activities, including the vitrification of high-activity sludge waste through the DWPF for eventual disposal in a federal repository for HLW, is the primary basis for the need to apply interim processing (DDA, ARP/MCU) prior to the SWPF coming on line in 2009. Preparation and removal of waste for disposal is expected to achieve risk-reduction through two primary means: reduction of high-activity waste stored in old-style tanks and reduction of the number of transfers and operations at the tank farm facility that could result in spills or accidental exposures to workers. Reducing the amount of waste stored in old-style tanks is expected to reduce risk because twelve tanks without secondary containment have a history of leakage; sufficient waste has been removed from these tanks such that there are currently no active leak sites. In addition, reducing tank inventory is expected to reduce worker risk because, when the tank farm is close to capacity, the number of operations and the complexity of the sequence of operations needed to manage the waste increases substantially (DNFSB, 2001).

In the RAI, NRC staff asked for additional information to support DOE's conclusion that the use of less efficient interim treatment measures before completion of the more efficient SWPF is consistent with the NDAA criterion of removal of highly radioactive radionuclides to the maximum extent practical. In response, DOE provided a comparison of risk-related metrics between three cases: a baseline case (DDA, ARP/MCU, SWPF), a case employing ARP/MCU and SWPF (no DDA), and a case employing SWPF only. DOE predicted the following five risk-related metrics for each case: dose to the public, dose to the inadvertent intruder, dose to the worker, risk in terms of old-style tank-years, and risk in terms of curie-years. Cost comparisons were provided, including the costs associated with slowing down DWPF waste processing. DOE also estimated the cost of shutting down the DWPF, but concluded that, in each case, it was preferable to slow down DWPF waste processing to avoid DWPF shutdown. Table 4 is a summary of the risk and cost information provided for the three cases considered.

Although the construction of the ARP/MCU facilities and the implementation of the DDA process will require resources, DOE expects that implementing interim treatment processes will reduce operational costs by reducing the number of years of operation of the HLW system (i.e., the tank farms, DWPF, SWPF, SPF, and SDF). DOE expects that deferring treatment until the ARP/MCU or the SWPF is operational will prolong the operation of the HLW system for two reasons. The first is simply that waste treatment will begin at a later time. The second is that the time required to treat waste at the SWPF will increase if interim processing is not performed because less tank space will be available to prepare waste for SWPF treatment and it will, therefore, take more time for the SWPF to process waste at its full capacity. The costs associated with performing limited interim processing (ARP/MCU and SWPF only) or no interim processing (SWPF only) are an additional $1.0 billion or $1.5 billion, respectively (2004 dollars). All-pathway public doses are not changed significantly because the all-pathway public doses are dominated by the radionuclides I-129 and Se-79 and the treatment processes were not designed to selectively remove these radionuclides. Intruder doses are lower in the cases without interim processing or with only limited interim processing than they are in the baseline case because, in both alternate cases, the SDF would not contain DDA waste. Intruder doses are expected to be higher if the SDF contains DDA waste because DOE predicts that Cs-137 dominates the dose to an inadvertent intruder (Cook, et al., 2005) and DDA waste has a higher Cs-137 concentration than either ARP/MCU or SWPF waste (d'Entremont and Drumm, 2005). In the limited interim treatment case and no interim treatment case, worker doses increase by 24% and 35% as compared to the baseline case, respectively. Because the tanks are aging, there is a risk associated with the potential loss of containment. In particular, some of the old-style tanks have developed leaks due to stress corrosion cracking at welds. DOE has a tank inspection program to evaluate the integrity of the tanks. In addition, where possible, the relative humidity within the tank vaults is limited to reduce the rate of general corrosion of the carbon steel. Because the probability of the development of leaks and the effects of DOE's tank inspection and maintenance programs are difficult to quantify, the risk of continued use of the old style tanks has been quantified with a metric of old-style tank years rather than estimates of the effects of potential leaks. This metric represents the cumulative amount of time that tanks without full secondary containment would contain waste under each treatment option. The limited interim processing case represents a 25% increase (to 300 tank-years) whereas the no interim processing case represents a 42% increase over the baseline case. While old-style tank years is not a direct measure of risk because practices are in place to minimize the future loss of containment of a tank or tanks, this metric does provide some indication of the potential for risk or the need for expensive alternatives to remediate the effects of a possible leak. The effects of each case on the life cycle of the entire tank farm are

Table 4 Summary of Risk and Cost Information for Three Waste Treatment Approaches (based on d'Entremont et al., 2005)

Metric	Baseline (DDA, ARP/MCU, SWPF)	Limited Interim Processing (ARP/MCU, SWPF)	No Interim Processing (SWPF)
Additional cost (millions, in 2004 dollars)	Not Applicable	1,000	1,500
Public dose - During operations (mrem/yr)	0.19	0.19	0.19
Intruder dose (mrem)	9	0	0
Public dose - After operations (mrem/yr)	2.3	2.3	2.3
Worker dose - total, all workers (rem)	890	1100	1200
Storage risk (old-style tank-years)	240	300	340
Storage risk (curie-years)	3.7×10^9	4.7×10^9	5.3×10^9

quantified in terms of curie-years, which is the sum of the number of years each tank is in service multiplied by the activity in the tank. Because waste treatment is deferred in the cases in which no interim processing or limited interim processing is performed, the number of tank farm curie-years increases to 25% greater than the baseline value in the limited interim processing case and 42% greater than the base line value in the case in which no interim treatment is performed.

In addition to evaluating cases in which limited or no interim processing was performed, DOE evaluated the possibility of creating the tank space necessary to support SWPF operation at full capacity by building additional tanks. Based on estimates developed for Hanford, DOE estimated that it would take approximately seven years and cost $300 million (2001 dollars) to construct four new tanks at SRS (WSRC, 2005b). DOE noted the $300 million cost of creating four new tanks would be approximately twice the remaining lifecycle cost of the ARP/MCU facility (i.e., lifecycle costs of the ARP/MCU facility less associated costs that have already been incurred) (d'Entremont et al., 2005). Furthermore, the $300 million cost of building new tanks without performing interim processing does not reflect the total cost associated with deferring processing until additional tanks could be built because it does not include costs associated with slowing down waste processing at the DWPF when tank space becomes limiting. In addition, DOE predicts that by the time the new tanks could be constructed and placed into operation, the SWPF will become operational, thereby eliminating the need for additional tank space. For these reasons, DOE concluded that building additional tanks would not be a practical way to alleviate the tank space shortage.

3.8 NRC Evaluation – Selection of Treatment Processes

To evaluate whether DOE planned to remove the identified highly radioactive radionuclides to the maximum extent practical, NRC staff evaluated information regarding technologies and processes available for waste treatment. As described in Section 3.7.1, WSRC, under contract to DOE in 1998, evaluated over 140 technologies to remove Cs-137 from SRS salt waste, and both DOE-SRS and DOE headquarters reviewed the results of this investigation. In addition, DOE headquarters requested and received input from NAS about the technologies selected and the selection process itself. Technologies were selected on the basis of expert judgment about the level of development of the technologies and the associated uncertainties in the technologies' ability to meet the waste treatment goals while minimizing potential impacts to worker safety and project schedules. NRC staff evaluated the original WSRC review of existing Cs removal technologies, the record of decision for selection of CSSX removal, and the NAS reviews of the technology selection and selection process, and concludes that the selection of CSSX removal technology appears to be appropriate.

In addition to Cs-137, DOE identified Sr-90 and four alpha-emitting isotopes of Pu, Am, Np, and Cm as highly radioactive radionuclides in the context of the Section 3116 draft waste determination for SRS salt waste. In both the ARP/MCU and SWPF processes, as described in Section 1.2, DOE intends to use cross-flow filtration to remove the insoluble fraction of Sr-90 and actinides and relies on sorption onto MST and subsequent filtration to remove the dissolved fraction of Sr-90 and actinides. DOE presented little information about the selection of the MST process. NRC staff recognizes that DOE may have based the selection of the MST sorption process on a thorough evaluation of Sr-90 and actinide removal processes that was not presented to NRC staff. The NAS 2000 and 2001 reviews (NAS, 2000; NAS, 2001) noted that the MST sorption technology appeared to be promising but had potential technical deficiencies that SRS staff were, at the time, working to resolve. In addition, the 2001 NAS report indicated that SRS staff were considering two alternatives to MST sorption for Sr-90 and actinide removal. Based on this information and the assumption that the high expected efficiencies of removal of Sr-90 and actinides by MST sorption with filtration that were reported by DOE (WSRC, 2005c) will be achieved, NRC staff concludes that the use of MST sorption with subsequent filtration to remove Sr-90 and actinides appears to be appropriate.

As noted in Section 3.5, the ARP facility can be used in a "filtration only" mode in which MST strikes are not used. DOE originally proposed to use MST as part of the ARP/MCU process only if additional actinide removal was necessary for the grouted waste to meet Class C limits (DOE, 2005). In response to a RAI from NRC asking for information to support the conclusion that using MST sorption only if required to meet Class C limits was consistent with removal of highly radioactive radionuclides to the maximum extent practical, DOE indicated MST sorption would be used for all batches of waste processed by the ARP/MCU procedure as long as the necessary processing throughput could be maintained (WSRC, 2005b). In addition, DOE indicated that criteria for the application of MST strikes would be developed as operational experience with the ARP/MCU process was gained and effects on operational throughput were better known. NRC staff believes it is appropriate to specify criteria for the application of MST sorption strikes as operational experience is gained because operational information is necessary to be able to compare the risks and costs of delaying treatment to the benefits of the additional amount of actinide removal that is accomplished with MST strikes. NRC staff concludes that operation of ARP/MCU without MST strikes will be acceptable if required by processing throughput requirements because (1) some actinide removal will be accomplished by filtration without MST strikes, (2) a relatively small amount of waste is expected to be processed by ARP/MCU as compared to the amount of waste to be processed by the SWPF, and (3) the use of MST is not necessary for consistency with DOE's evaluation of the class of the waste or the SDF's ability to meet the performance objectives because these outcomes were calculated based on the assumption that MST strikes would not be used during ARP/MCU treatment.

As discussed in Section 3.7.1, DOE considered technologies for deliquification and solids separation to be used in the DDA process. In addition, DOE presented information about the selection of tanks to be treated with DDA before ARP/MCU and the SWPF are operational. NRC evaluated information regarding the technologies that will be used in the DDA process as well as the criteria used to select tank waste that will be treated with DDA. Because the DDA process is the only proposed process that does not use filtration and is, therefore, not expected to achieve approximately 100% removal of insoluble radionuclides, it is the only process for which the removal efficiencies will be affected significantly by the amount of sludge entrained in dissolved salt waste. Because the predicted radionuclide removal efficiencies of the ARP/MCU and SWPF demonstrate that removal of Cs-137, Sr-90, and actinides can be accomplished to a greater degree with different treatment technologies, the use of technologies used in the DDA process must be considered in terms of DOE's two-phase, three-part process. NRC staff concludes that the selection of waste with low supernate activity and relatively small volumes of low-heat waste sludge, in addition to the treatment of a relatively small amount of waste with the DDA process, are important to the acceptability of the DDA interim process. NRC staff notes that, because a relatively small amount of waste will be treated by DDA, the overall removal of most radionuclides from the salt waste is driven by removal in the SWPF (Table 3). NRC staff also notes that, while the removal of soluble radionuclides that are significantly removed by downstream processes (e.g., Cs-137) is affected by the extent of deliquification achieved during DDA processing, the removal of other soluble radionuclides is not affected by the degree of deliquification because, whether or not they are separated from the saltcake and processed at the SWPF, they ultimately are sent to the SDF (e.g., Tc-99, I-129).

In response to the action items (WSRC, 2005a), DOE indicated that it did not target Se-79, Tc-99, or I-129 for removal because DOE predicted those radionuclides would cause a small dose to the public. However, as discussed in Section 3.4, the NRC staff has concluded that DOE has provided insufficient support for its base case to be able to discount the potential dose contributions of Se-79, Tc-99, and I-129 (also see Section 4). NRC concludes that DOE's decision to include Se-79, Tc-99, and I-129 as highly radioactive radionuclides is appropriate and that Criterion Two requires removal of these radionuclides to the maximum extent practical. DOE concluded that removal of the dissolved phase of Se-79, Tc-99, or I-129 would be impractical because treatment technologies were immature. To support the conclusion that no economically practical alternatives were available to remove Se-79, Tc-99, or I-129 from salt waste, DOE cited an evaluation of the feasibility of removal of various radionuclides from Hanford site tank waste that was completed in 1996 (WHC, 1996). The cited document identified technologies to remove both dissolved Se-79 and Tc-99 that had been tested at the bench scale, but omitted at least one technology that had successfully removed Tc-99 from simulated and actual decontaminated SRS salt waste in bench-scale tests (WSRC, 1994). The cited document indicated that, at the time it was completed, there were no technically practical technologies available to remove Se-79, Tc-99, or I-129 from Hanford tank waste. Based on the information provided by DOE, and DOE's statement that it has not performed significant research and development activities regarding removal of these radionuclides, NRC concludes that implementation of additional treatment technologies would not be practical within the constraints imposed by DOE's proposed schedule. Because DOE indicated that it did not target Se-79, Tc-99, or I-129 for removal, NRC's conclusion that additional removal would not be practical is not based on DOE's assessment of existing technologies, but rather on consideration of the impacts of delays to the proposed schedule that would result from changes to the treatment process design to implement applicable technologies. As discussed in Section 3.7.2, DOE has indicated that delays to the proposed schedule would cause significant increases in project cost and worker risk. NRC concludes that the proposed removal of Se-79, Tc-99, and I-129 is consistent with removal to the extent practical because, even if DOE were aware of technologies that could remove dissolved Se-79, Tc-99, or I-129 from salt waste, delays associated with implementing these technologies at this point in the design of the treatment processes would cause unjustified increases in project cost and worker risk. Because the conclusion is based on the impacts of delays caused by implementation of additional treatment technologies, any changes to the proposed schedule that would allow applicable technologies to be developed or implemented without causing additional delays would be expected to impact this conclusion.

The waste that will receive the least amount of treatment before being sent to SDF is the waste in Tank 48. Although the Tank 48 waste represents only 0.3% of the volume of the salt waste, it represents 27% of the activity projected to be disposed of in the SDF. DOE was unable to develop a practical option for treatment of the Tank 48 waste, other than directly processing it in the SPF and disposing of the resultant material in the SDF. Because Tank 48 contains a relatively small amount of activity (3×10^{16} Bq [0.8 MCi]) compared to other tanks that could be emptied to create as much space, disposing of Tank 48 waste in the SDF creates a relatively large amount of new-style tank space per Ci sent to SDF. DOE evaluated alternatives for treating other tank wastes with DDA processing and found that creation of a comparable amount of tank space by treating waste in an alternate tank with DDA processing would result in a greater activity of waste being sent to the SDF. In addition, because of its location, Tank 48 is the planned feed tank for the ARP/MCU process and is an integral part of the plan for staging feed for the SWPF. Based on the following factors, NRC staff concludes that there is reasonable assurance that Tank 48 waste has been processed to the maximum extent practical: (1) the Tank 48 waste represents a small fraction (0.4% by activity) of the total waste in or projected for the tank farm facilities, (2) other treatment options were considered and were determined to not be technically practical, (3) Tank 48 is a new style tank containing a relatively small amount of waste (3×10^{16} Bq [0.8 MCi]) and has a strategic location with respect to tank farm operations and, (4) DOE estimated that the saltstone resulting from processing Tank 48 waste would result in a wasteform that meets Class C concentration limits (see Section 4.1.1).

In addition to evaluating the selection of processes to remove radionuclides from salt waste, NRC staff also evaluated information regarding alternatives to DOE's proposed two-phase, three-part approach. Because SWPF will be able to remove a greater fraction of most highly radioactive radionuclides than either DDA or ARP/MCU, NRC requested information to support DOE's conclusion that using DDA and ARP/MCU rather than treating all of the waste with the SWPF is compatible with the criterion for removal of highly radioactive radionuclides to the maximum extent practical.

In its draft waste determination, DOE stated that removal of radionuclides "to the maximum extent practical" will vary depending not only on technologies but also the overall costs and benefits, and allows consideration of factors such as environmental, health, timing or other exigencies, considerations that may ensue from delay of waste removal, and reasonable availability of proven technologies. NRC staff agrees that multiple factors can and should be considered when weighing what is the maximum extent practical for any given situation. In the 1993 petition denial establishing its WIR criteria, the NRC stated that, with regard to decommissioning of reprocessing facilities, both economic factors and technical factors could be considered, as long as there is adequate protection of public health and safety. In previous incidental waste reviews performed for DOE, NRC reviewed information on monetary costs as well as worker doses (NRC, 2003a). In its RAI for this draft waste determination, staff requested additional quantitative information to support DOE's assertion that the proposed approach would remove highly radioactive radionuclides to the maximum extent practical.

In response to the RAI, DOE performed an analysis of the economic and mission impacts of alternative approaches in which waste processing would be postponed until either the ARP/MCU processes or the SWPF becomes operational. DOE analyzed a "limited interim processing" case in which the DDA process would not be used and a "no interim processing" case in which neither DDA nor ARP/MCU interim processes would be used.

Because the radionuclide removal efficiencies achieved by the interim processes are lower than the predicted radionuclide removal efficiencies that will be achieved by the SWPF, use of interim processes will result in higher radionuclide concentrations and total activity in the SDF than cases in which waste treatment is delayed until the SWPF becomes operational. The increased future risk to the public due to the disposal of greater quantities of highly radioactive radionuclides in the SDF must be evaluated in the context of the risks, costs, and mission impacts associated with cases in which interim processing is not performed. There are two types of risks associated with deferring waste processing until the SWPF becomes operational. The first is the risk to the public associated with the possibility that a leak will develop in an old-style tank. It is difficult to estimate the risk to the public associated with a leak in a HLW tank because it is difficult to predict the probability that a leak will develop, and because, if a leak were to be detected, it is likely that mitigative measures would be used to limit the risk to the public. The second type of risk is risk to facility workers. Cumulative radiological risk to facility workers is expected to increase if waste treatment is deferred because the tank farm would operate for a longer period of time (Table 4) (d'Entremont et al., 2005). In addition, if less space is available in the tank farm to prepare waste for DWPF processing, radiological risk to workers is expected to increase because the number of tank-to-tank transfers necessary to maintain operation of the DWPF would increase (DNFSB, 2001).

Thus, the primary considerations as to whether employing interim processing technologies that are less efficient while waiting for the more efficient technologies to become available in achieving removal of highly radioactive radionuclides to the maximum extent practical are cost and risk transferral. Employing the interim processing is expected to reduce worker risk, while increasing future risks to an inadvertent intruder (Table 4). The risk to a member of the public that does not intrude on the site is not expected to change if waste treatment is deferred until the SWPF is operational (Table 4) because the predicted dose to a member of the public in DOE's base-case PA is dominated by Se-79 and I-129 (see Sections 4.2.15 and 4.2.16), which are expected to be removed to a similar extent by the interim processes and by the SWPF. It is difficult to compare worker risk to public risk because worker risk is accepted by the individual while public risk is imposed.

The additional costs associated with deferring waste treatment until the SWPF becomes operational are easier to quantify. As discussed in Section 3.7.2, DOE expects that deferring treatment until the ARP/MCU or SWPF become operational will increase the total costs of treating the waste primarily because of the added operational costs of operating the HLW system for additional time. DOE predicts that the cost associated with the baseline option is significant ($4.9 billion in 2004 dollars), but that the cost associated with performing only limited interim processing (i.e., use of ARP/MCU but no use of DDA) ($5.8 billion in 2004 dollars) and the costs associated with postponing waste treatment entirely until the SWPF becomes operational ($6.3 billion in 2004 dollars) would be even greater (Table 4). Therefore, if DDA processing is not used, dose to an inadvertent intruder is expected to decrease by 0.09 mSv/yr (9 mrem/yr) at a cost of approximately $1.0 billion (2004 dollars) while worker risk is increased by 210 person-rem. If no interim processes are used, dose to an inadvertent intruder is expected to decrease by 0.09 mSv/yr (9 mrem/yr) at a cost of approximately $1.5 billion dollars (2004 dollars) while worker risk is increased by 310 person-rem. Because the NRC staff believes there is reasonable assurance that worker and public performance objectives will be satisfied if DDA and ARP/MCU are used to process the volume of low-activity waste described in the draft waste determination and supporting documents (d'Entremont and Drumm, 2005), NRC staff concludes that expenditure of an additional $1.0 billion (2004 dollars) to $1.5 billion (2004 dollars) to decrease public exposures further below the performance objectives would not be warranted (see Sections 4.2.16 and 4.2.18).

Use of the interim processes to treat more waste or higher activity waste than proposed would be expected to increase the inventory of highly radioactive radionuclides in the SDF, which would be expected to affect the demonstration that the performance objectives can be met. Because the demonstration that the use of the interim processes is consistent with the criterion of removal of radionuclides to the maximum extent practical depends in part on the use of the interim processes to treat a relatively small volume of the lowest activity waste, use of the interim processes to treat more or higher activity waste than proposed would be expected to impact whether Criterion Two can be met.

Because DOE predicts the primary worker risk reduction and cost savings achieved by implementing interim waste treatment processes result from reducing the number of years of operation of the HLW system (d'Entremont et al., 2005), changes to the proposed waste treatment schedule would be expected to change the cost/benefit analysis used to support the conclusion that highly radioactive radionuclides will be removed from the salt waste to the maximum extent practical. In addition, because schedule constraints were cited as the primary reason for some of DOE’s technology selections, it appears that changes to the proposed schedule could affect DOE’s choice of salt waste treatment technologies.

3.9 NRC Review and Conclusions (Criterion Two)

The NRC staff concludes that Criterion Two will be met for SRS salt waste. This conclusion is based primarily on the NRC staff’s four intermediate conclusions, as follows: (1) DOE used an appropriate process to identify highly radioactive radionuclides in the context of the Section 3116 draft waste determination for SRS salt waste (Section 3.4), (2) relatively high removal efficiencies are predicted for most of the highly radioactive radionuclides in SRS salt waste (Section 3.6), (3) appropriate processes were used to evaluate technologies for the treatment of Tank 48 waste and additional removal of radionuclides from the waste is not technologically or economically practical (Sections 3.7.1 and 3.8), and (4) the additional costs and radiological risks to workers associated with delaying salt waste treatment until the ARP/MCU or SWPF is operational do not appear to be justified by the small reduction in the risk to an inadvertent intruder and the public that could be achieved by delaying treatment (Section 3.8). As discussed in Section 3.8, the NRC staff agrees with DOE's position that it is appropriate to consider multiple factors when evaluating the practicality of radionuclide removal in the context of Criterion Two. For example, although DOE predicts that very low efficiencies of removal of I-129 and Tc-99 will be achieved, the NRC staff concludes that the removal of these radionuclides is not inconsistent with Criterion Two because implementation of additional treatment technologies was determined not to be practical within the constraints of DOE's proposed schedule. Similarly, NRC staff considered DOE's need to create space in new-style tanks when evaluating DOE's selection of waste treatment technologies and its proposed two-phase, three-part approach. The primary benefit of using interim processes, as identified by DOE, is that these processes will create enough compliant tank space to support operation of the SWPF at close to its full capacity when it becomes operational. Significant changes in the proposed schedule would be expected to impact the evaluation of which technologies could be implemented in time to create the tank space necessary to support SWPF start up, as well as the relative costs and benefits of implementing various treatment technologies. In addition, the NRC staff notes that the conclusion that the proposed two-phase, three-part approach is consistent with Criterion Two is based in part on the use of the interim processes to treat a relatively small volume of the lowest activity waste. Use of the interim processes to treat more waste or higher activity waste than proposed would be expected to affect both the demonstration that the performance objectives can be met and the demonstration that Criterion Two can be satisfied.

<u>NRC staff used the following assumptions in assessing conformance with Criterion Two</u>:

1. The DDA and ARP/MCU interim treatment processes will not be used to treat significantly more waste or higher activity waste than proposed (see Section 3.8).

2. The actual radionuclide removal efficiencies of the ARP/MCU and SWPF treatment processes will meet or exceed the reported lower bounds of the projected removal efficiencies (see Sections 3.6 and 3.8).

3. During the DDA process, dissolution of saltcake and solids settling will be performed such that the amount of sludge entrained in salt waste after DDA processing does not exceed DOE's estimate of 200 mg/L salt waste, which was identified by DOE as being a conservative estimate (WSRC, 2005c) (see Section 3.6).

4. During DDA processing, deliquification will be performed such that the lower bound of predicted deliquification (30%) (WSRC, 2005c) is met or exceeded (see Section 3.6).

5. Financial data used to support DOE's cost estimates are reasonable (see Section 3.8).

6. The proposed dates of operation of the ARP/MCU and SWPF facilities will be met (see Section 3.8).

7. The Tank 48 waste can be processed safely in the SPF, and the wasteform will meet the Class C limits and perform as expected (see Section 3.8).

NRC staff concludes the following with respect to Criterion Two:

1. DOE's conclusion that highly radioactive radionuclides will be removed to the extent practical by the proposed two-phase, three-part process, including the proposed process for the management of Tank 48 waste, is reasonable.

4. CRITERIA THREE (A) AND THREE (B)

(3) (A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, and will 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; and

(ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or

(B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will 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;

(ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and

(iii) pursuant to plans developed by the Secretary in consultation with the Commission

Whether Criterion 3(A) or Criterion 3(B) is applicable depends on the class of the waste being assessed in the waste determination. In the case of the draft waste determination for salt waste disposal at SRS, Criterion 3(A) is applicable because the final waste form disposed of in the SDF is not expected to exceed Class C concentrations (see Section 4.1). Therefore, Criterion 3(B) is not evaluated in this TER. NRC staff also did not assess 3(A)(ii) because the State of South Carolina has authority over State-approved closure plans and State-approved permits.

Criterion 3(A) refers to the performance objectives of 10 CFR Part 61, Subpart C, which require assessment of protection of the general population from releases of radioactivity, protection of individuals from inadvertent intrusion into the waste, protection of individuals during operations, and evaluation of the stability of the disposal site after closure. Protection of the general population (including intruders) is typically evaluated through a PA calculation that takes into account the relevant physical processes and the temporal evolution of the system. The NRC staff's assessment of DOE's PA is presented in Section 4.2.

4.1 Assessment of Waste Classification

(3)(A) (The waste) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations,

4.1.1 Waste Classification

10 CFR 61.55 classifies low-level radioactive waste intended for near surface disposal as Class A, B, or C. Table 5 provides a comparison of the radiological composition of the average waste resulting from the DDA, ARP/MCU, and SWPF processes with the Class C concentration limits found in Tables 1 and 2 in 10 CFR 61.55. The Class C concentrations in Tables 1 and 2 in 10 CFR 61.55 were not provided for every radionuclide, only the dominant radionuclides expected in a commercial low-level waste stream. The saltstone wasteform will contain a mixture of long- and short-lived radionuclides, and therefore 10 CFR 61.55 (a)(5) was applied to determine waste classification. 10 CFR 61.55 (a)(5) states:

> "Classification determined by both long- and short-lived radionuclides. If radioactive waste contains a mixture of radionuclides, some of which are listed in Table 1, and some of which are listed in Table 2, classification shall be determined as follows: (i) If the concentration of a nuclide listed in Table 1 does not exceed 0.1 times the value listed in Table 1, the class shall be determined by the concentration of the radionuclides listed in Table 2. (ii) If the concentration of a nuclide listed in Table 1 exceeds 0.1 times the value listed in Table 1 but does not exceed the value in Table 1, the waste shall be Class C, provided the concentration of nuclides listed in Table 2 does not exceed the value shown in Column 3 of Table 2."

The waste treated through interim processing and the SWPF is estimated to meet the concentration limits established in 10 CFR 61.55 for Class C waste as shown in Table 5. The wasteform produced from waste processing is a formulation of cement, fly ash, and blast furnace slag mixed with waste in an approximate ratio of 1:1 to produce a solidified wasteform to achieve stability of the waste for disposal (see Section 4.2.6). The waste and solidifying materials will be mixed during processing to achieve reasonable homogeneity. The NRC Branch Technical Position on concentration averaging is being applied by DOE to calculate the average concentration of the final wasteform (NRC, 1995).

4.1.2 NRC Evaluation - Waste Classification

In developing the waste classification system in 10 CFR 61, consideration was given to long-lived radionuclides whose potential hazard will persist long after institutional controls, improved waste form, and deeper disposal have ceased to be effective and short-lived radionuclides for which the requirements on institutional controls, waste form, and disposal methods are effective. Tables 1 and 2 of 10 CFR 61.55 provide the concentration limits for classifying low-level radioactive waste intended for near surface disposal as Class A, B, or C. The waste classification system was originally developed primarily to facilitate waste generators' assessment as to whether their material was suitable for packaging and transport to a waste disposal facility, and secondarily to identify technical requirements for a waste disposal facility developer. The 10 CFR 61 classification system is based on the concentration of radionuclides

Table 5 Comparison of DDA, ARP/MCU, and SWPF Waste with Class C Concentration Limits [d'Entremont and Drumm, 2005][1]

Radionuclide	Class C Limit (Ci/m^3)	DDA Waste (Ci/m^3)	ARP/MCU Waste (Ci/m^3)	SWPF Waste (Ci/m^3)
Long-Lived				
C-14	8	0.001	0.001	0.0008
Tc-99	3	0.006	0.03	0.06
I-129	0.08	0.000004	0.00001	0.00003
Alpha Emitting Transuranic (TRU) nuclides with half-life greater than 5 years [2]	100	41	2	<10
Pu-241 [2]	3500	2	0.1	7
Cm-242 [2]	20000	0.00005	0.00008	0.0001
Short-Lived				
Ni-63	700	0.002	0.0009	0.0002
Sr-90	7000	0.1	0.02	0.002
Cs-137	4600	22	8	0.005

[1] Values in table are different from those in the draft waste determination (March 2005) due to updated values provided in d'Entremont and Drumm (June 2005).
[2] Units are nanocuries per gram

in waste and is intended to limit exposures to the inadvertent intruder, whereas protection of the general population is addressed via limits on the total radionuclide activity disposed of at a facility. It is expected that a waste disposal facility would provide a demonstration of how its disposal facility design can achieve the Subpart C performance objectives for the expected waste stream. It would not be unexpected that there may be radionuclides, not listed in Tables 1 or 2, that are present in a waste stream that cause a significant risk. Likewise, because different disposal techniques or systems are possible, waste that does not meet Class C concentration limits can be determined to be suitable for near surface disposal by applying the alternate waste classification criteria found at 10 CFR 61.58. Therefore satisfaction of the performance objectives is a necessary element to achieving safety whether Class C limits are satisfied or not (see Section 4.2.10).

Table 5 shows the expected average radionuclide concentrations for the DDA, ARP/MCU, and SWPF waste streams. The waste is expected to contain both short- and long-lived radionuclides, therefore application of 10 CFR 61.55 (a)(5) is appropriate for determining waste classification. The sum of fractions for the long-lived radionuclides is 0.41, 0.03, and 0.10 for the DDA, ARP/MCU, and SWPF waste streams, respectively. The sum of fractions for the short-lived radionuclides is 0.005, 0.002, and 2×10^{-6} for the DDA, ARP/MCU, and SWPF, respectively. The long-lived radionuclides are more limiting in defining the waste classification, although all waste streams are expected to be Class C or lower. Although not expected to occur, uncertainty in radionuclide concentrations could result in some batches of DDA waste not meeting Class C concentration limits (see Section 3.2).

DOE has appropriately applied the NRC Branch Technical Position on concentration averaging for the saltstone wasteform (NRC, 1995). The wasteform produced from waste processing is a formulation of cement, fly ash, and blast furnace slag mixed with waste in an approximate ratio of 1:1 to produce a solidified wasteform to achieve stability of the waste for disposal. Based on the information provided by DOE, the NRC staff agrees that the waste and solidifying materials will be mixed during processing to achieve reasonable homogeneity. Although there is uncertainty in the characterization data, the NRC staff believes the waste treated through interim processing and the SWPF will likely meet the concentration limits established in 10 CFR 61.55 for Class C waste as shown in Table 5.

4.2 Performance Assessment to Demonstrate Conformance with Performance Objectives

Typically a PA is developed to demonstrate whether the performance objectives have been met. A PA is a quantitative evaluation of potential releases into the environment and the resultant radiological doses. Depending on the computational tools used by the analyst, the PA may be a single integrated model, or it may represent an analysis approach for integrating and evaluating a collection of other models. Performance assessments involve the integration of process models to identify and propagate impacts and uncertainties between models. Process models are used to evaluate physical and chemical phenomenon such as release of radionuclides from wasteforms, degradation of engineered components, and transport of radionuclides through environmental media, among others. The PA documentation will commonly provide the justification for the data used, a description of the models used, verification of and support for the models, and an evaluation of the impact of data and model uncertainty. To evaluate uncertainty, a variety of techniques typically are used, including deterministic analysis with sensitivity analysis, and probabilistic analysis with uncertainty and sensitivity analyses. The sensitivity analysis may then be used to conduct a risk-informed evaluation through the in-depth review of those parameters and processes most important to system performance in relation to meeting the performance objectives. The following TER sections provide a discussion of the main components of DOE's PA to demonstrate compliance with the performance objectives for the saltstone waste disposal facility.

In general, all approaches to PA calculations (e.g., deterministic, probabilistic) have their advantages and disadvantages. A deterministic approach can be very valuable when the analysis is clearly conservative because it makes the demonstration of meeting the performance objectives more straightforward and it can be significantly easier to interpret results and explain them to stakeholders. While deterministic analysis can be a suitable methodology for PA, it can also present a challenge for a system that responds in a highly nonlinear fashion with changes in the independent variables. In addition, when there are numerous inputs (e.g., data or models) that are uncertain, the evaluation of the impacts of the uncertainties on the decision can be difficult. Typical one-off type of sensitivity analysis where a single parameter is increased or decreased will only identify local sensitivity within the parameter space, such that it may not clearly identify the risk implications. A probabilistic approach can have distinct advantages when there are a number of uncertainties that may significantly influence the results of a PA.

The term "conservatism," as used with respect to PA, is a relative term that needs to be placed within the proper context. Conservatism is typically defined mainly with respect to what is known or sometimes with standard practices that have been demonstrated to yield acceptable performance. In this regard, model support (i.e., information that supports the results of a model) of process model results plays a key role in developing confidence in the output of PA calculations. Because of the long time periods involved with the analysis, PA models cannot be validated in the traditional sense. However, multiple methods for developing confidence in the model projections can be used, including: laboratory experiments, alternative modeling approaches, field measurements, natural analogs, and expert elicitation, among others. The amount of model support provided should be commensurate with the risk reduction being provided by the natural and engineered system. Multiple lines of evidence are strongly encouraged when the risk reduction of the systems being evaluated is large.

4.2.1 Performance Assessment Overview

The Radiological Performance Assessment (Cook and Fowler, 1992) for the SDF was originally prepared to fulfill the requirements of DOE Order 5820.2A in 1992. This DOE order has since been superseded by DOE Order 435.1, Radioactive Waste Management. The 1992 PA was followed by an addendum in 1998 (Fowler, 1998) to address and incorporate comments from the DOE Performance Assessment Peer Review Panel and DOE headquarters. In 2002, a special analysis (Cook, et al., 2002) was performed to update the 1992 PA based on new information from the construction phase of the SDF and to provide the compliance determination for DOE Order 435.1 (DOE, 1999a, 1999b, 1999c). Using what it considered the latest information on the SDF feed solutions, updated modeling methods, and updated closure cap design and evaluations, DOE developed the 2005 special analysis (Cook, et al., 2005) as the latest iteration in the PA process. As stated by DOE in Rosenberger, et al. (2005), the 2005 special analysis supplements the analyses in the 1992 PA and supersedes the analyses of the 2002 special analysis.

To evaluate the long-term performance of the SDF in the 1992 PA, DOE developed site-specific conceptual models to consider exposure pathways and scenarios of potential importance, potential releases from the facility to the environment, and potential doses to a receptor as a result of exposure to radioactive contaminants. In the 1992 PA, DOE assumed that the inventory (source term) was composed of the decontaminated salt solution stream from the In-Tank Precipitation Facility and the concentrate stream from the Effluent Treatment Process (Cook and Fowler, 1992). The 1992 PA evaluated waste material in SDF Vaults 1 and 4 and projected the results for up to 14 future vaults. The groundwater pathway was the primary quantitative analysis for post-closure performance, and a qualitative evaluation concluded that the air pathway was insignificant in comparison. The resident farmer intruder scenario was evaluated qualitatively in the 1992 PA. The PORFLOW code developed by Analytical and Computational Research Inc. was the primary computer code used to evaluate the groundwater pathway (Runchal and Sagar, 1991). PORFLOW was used to simulate groundwater flow and contaminant transport and to calculate maximum concentrations of radionuclides in groundwater at locations beyond the 100-m (328-ft) buffer zone used in the 1992 PA. Estimates of dose from direct ingestion of contaminated groundwater were then made by multiplying the radionuclide concentrations by a radionuclide-specific dose conversion factor of annual dose per unit concentration. The 1992 PA concluded that the groundwater dose calculations were well within the performance limits of DOE and therefore an all-pathways evaluation was not completed because DOE had reasonable assurance that the all-pathways dose would be within the performance limits (Rosenberger, et al., 2005).

The 2005 special analysis (Cook, et al., 2005) was intended to evaluate the SDF against the specific performance objectives of DOE Order 435.1 and 10 CFR Part 61, Subpart C in an all-pathways analysis. The 2005 special analysis also was intended to update the 1992 PA by using the current estimates of total inventory expected to be disposed of in Vault 4 and to recalculate the groundwater and air transport exposure pathways (Rosenberger, et al., 2005). The all-pathways analysis included residential and agricultural pathways (i.e., resident farmer scenario) and sensitivity evaluations. The 2005 special analysis did not alter the 1992 PA conceptual model of the migration of radionuclides from the saltstone wasteform in Vault 4 to the environment via diffusion and advection (Cook, et al., 2005).

The 2005 special analysis included 45 radionuclides (and their progeny) selected with a screening procedure recommended by the National Council on Radiation Protection and Measurements (NCRP) (NCRP, 1996), but only addressed the inventory projected for Vault 4. The PA methodology is summarized as follows. The groundwater pathways analysis used a sequence of three computer codes. The Hydrologic Evaluation of Landfill Performance (HELP) code [U.S. Environmental Protection Agency (EPA), 1994a,b] was used to calculate water infiltration over time through the upper geosynthetic clay layer. The PORFLOW code (Analytical and Computational Research, Inc., 2002) was then used to calculate groundwater transport over time below the upper geosynthetic clay layer and in the saturated zone, as well as to calculate dissolved radionuclide fluxes to and within the saturated zone to the 100-m (328-ft) well location. The third step in the process uses the LADTAP XL© spreadsheet code (Simpkins, 2004) to calculate doses to the resident farmer from contaminated groundwater use at 100 m (330 ft) down-gradient through a number of residential and agricultural exposure pathways.

The air pathways exposure analysis was performed independently of the groundwater pathway analysis and used a sequence of two computer codes. The PORFLOW code (Analytical and Computational Research, Inc., 2002) was used to calculate the gaseous radionuclide diffusion from the vault to the ground surface. The EPA Clean Air Act Assessment Package-1988 (CAP88) code (EPA, 2002) was then used to calculate airborne transport and the resultant dose from possible air pathways (e.g., direct plume shine, inhalation, and ingestion of vegetables, meat, and milk exposed to the airborne radioactivity). Eight radionuclides were considered for the air pathway analysis (Rosenberger, et al., 2005). Doses from the air and groundwater pathways were then summed for each radionuclide to provide an all-pathways dose for comparison to the 10 CFR Part 61, Subpart C, performance objective.

Inadvertent intruder analysis was conducted quantitatively in the PA presented in the 2005 special analysis. A resident intruder scenario was evaluated using a software tool developed by DOE specifically for the inadvertent intruder evaluation at the Savannah River Site (Koffman, 2004). The software tool calculated the radionuclide-specific concentrations that would be encountered by an intruder by disregarding leaching and only considering decay for the amount of contaminant remaining at the site at the time of intrusion (Cook, et al., 2005).

4.2.2 Source Term

The source term, as defined here, is the radiological composition and quantity of material to be disposed of in the SDF. DOE estimates that the salt waste treatment and processing strategy will result in 6.27×10^8 L (170 Mgal) of grout containing 1.1×10^{17} Bq (3 MCi) of radioactivity to be disposed of in the SDF. The SDF has established WAC to ensure that the concentrations of waste accepted by the facility is suitable for disposal. The saltstone facility WAC establishes the radionuclide limits for the waste. A key element of DOE's approach to managing uncertainties associated with development of the source term for the PA calculations is to sample the material sent to the feed tank prior to transfer to the SPF.

In response to the RAI, DOE provided the sampling plan for the saltstone feed tank (Ketusky, 2005). The plan was developed prior to and during the development of the WAC. DOE intends to develop a Waste Compliance Plan (WCP), in which a detailed sampling plan will be finalized. Tank 50 is envisioned to be the receipt tank for waste to be transferred to the SPF. Possible waste stream influents include but are not limited to: Effluent Treatment Process (ETP) effluents, DWPF recycle, DDA solution, and post treatment dissolved salt solution from the ARP and MCU. Three main types of samples are to be obtained from Tank 50, namely: DDA pull sample, Grout pull sample, and operations (OPS) pull sample. Samples are to be taken from Tank 50 for a variety of uses, including: vault classification, grout qualification and formulation, verification that the grout will be nonhazardous, verification that the grout will meet processing and quality requirements, and compliance with liquid/solid chemistry and radiological contents requirements. The term pull sample is used to describe a bulk sample of up to 4 liters of material that then may be used for a variety of analyses. Sample frequency is from 1 per 5 years for a DDA pull sample to quarterly for Grout and OPS pull samples.

Material balance will be used to maintain the waste in Tank 50 compliant with the WAC. The material to be transferred to the SPF will be sampled prior to transfer to Tank 50 (as applicable). The characterization of the material transferred to the SPF will be maintained by the material balance. DOE indicated that process knowledge can be used in numerous aspects of the sampling strategy. In particular, based on the proposed permit with the State of South Carolina, WAC samples would not need to be taken from Tank 50 prior to sending material to the SPF.

The source term used in DOE's current PA calculations is based on estimates of the projected radiological inventory. The source term estimates have undergone considerable modification since the original PA was developed in 1992 as a result of new characterization information, new techniques to calculate concentrations of some radionuclides, and a new processing strategy to treat the waste. Table 6 provides the estimated total inventory (in curies) to be disposed of in the SDF for each radionuclide (Rosenberger et al., 2005, pg. 17). The total volume of material to be disposed of is 6.4×10^5 m^3 (2.26×10^7 ft^3). Section 3.1 provides a description of the approach to estimating the waste composition prior to disposal in the SDF.

DOE's approach to ensuring that the waste can achieve the performance objectives is to use the PA to develop inventory limits. The expected waste inventory (product of volume and radiological concentration of the waste) is compared to the inventory limits and a sum-of-fractions is calculated. If the sum-of-fractions is less than one, then the waste can satisfy the performance objective. As indicated above, sampling of the waste feed material prior to

Table 6 Estimated Source Term for the Saltstone Disposal Facility

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H-3	9.43×10^{3}	Ra-228	1.04×10^{-1}
C-14	5.20×10^{2}	Ac-227	1.91×10^{-5}
Na-22	5.05×10^{3}	Th-229	7.53×10^{-3}
Al-26	2.35×10^{1}	Th-230	3.53×10^{-2}
Ni-59	2.85	Pa-231	5.32×10^{-5}
Co-60	1.10×10^{2}	Th-232	1.04×10^{-1}
Ni-63	2.51×10^{2}	U-232	3.09×10^{-2}
Se-79	8.94×10^{1}	U-233	2.22
Sr-90	7.43×10^{3}	U-234	7.72
Y-90	7.43×10^{3}	U-235	1.35×10^{-1}
Nb-94	4.22×10^{-3}	U-236	3.03×10^{-1}
Tc-99	3.31×10^{4}	U-238	5.19
Ru-106	2.28×10^{3}	Np-237	2.12
Rh-106	2.28×10^{3}	Pu-238	1.36×10^{4}
Sb-125	9.24×10^{3}	Pu-239	6.55×10^{2}
Te-125m	2.26×10^{3}	Pu-240	1.75×10^{2}
Sn-126	4.51×10^{2}	Pu-241	7.03×10^{3}
Sb-126	6.30×10^{1}	Pu-242	1.81×10^{-1}
Sb-126m	4.50×10^{2}	Am-241	9.50×10^{1}
I-129	1.80×10^{1}	Am-242m	5.27×10^{-2}
Cs-134	2.71×10^{3}	Pu-244	7.96×10^{-4}
Cs-135	4.67	Am-243	2.18×10^{-2}
Cs-137	1.35×10^{6}	Cm-242	1.05×10^{-1}
Ba-137m	1.28×10^{6}	Cm-243	2.67×10^{-2}
Ce-144	6.27	Cm-244	8.72×10^{1}
Pr-144	6.27	Cm-245	8.58×10^{-3}
Pm-147	4.14×10^{3}	Cm-247	5.15×10^{-12}
Sm-151	4.55×10^{3}	Cm-248	5.36×10^{-12}
Eu-152	2.20×10^{1}	Bk-249	6.31×10^{-19}
Eu-154	9.74×10^{2}	Cf-249	4.79×10^{-11}
Eu-155	2.57×10^{2}	Cf-251	2.47×10^{-1}
Ra-226	1.30×10^{1}	Cf-252	5.32×10^{-14}
		Total	$\mathbf{2.74 \times 10^{6}}$

transfer to the SPF or calculation of the radiological waste composition will be used to perform the sum-of-fractions calculations for the relevant performance objectives.

4.2.3 NRC Evaluation – Source Term

NRC staff evaluated the information supporting DOE's estimated inventory. Cs-137 and its short-lived daughter Ba-137m are arguably the most well-characterized radionuclides in the tank farm waste. DOE estimated that the total activity of Cs-137 to be disposed of at the SDF would be approximately 1.1×10^{17} Bq (3 MCi), but could possibly range as high as 1.9×10^{17} Bq

(5 MCi) (DOE, 2005). A direct estimate of the uncertainty in the other radionuclides was not provided, but it would be expected, on a relative basis, to be larger as a result of more limited characterization and in some cases more complicated partitioning between the physical components of the waste (e.g., supernate, saltcake, sludge).

As indicated in Section 3.1, the estimation of tank inventory (which directly influences the eventual SDF inventory) and the radiological composition of each physical phase, is complicated by a number of factors. A number of inputs used to develop the tank farm inventory are based on assumptions, process knowledge, or other forms of indirect information. Even with the 2003 and 2004 campaigns to improve the estimates of the radiological composition of the supernatant and salt phases, the sampling of a number of highly radioactive radionuclides is fairly limited. Therefore there is uncertainty in the radiological composition of the waste feed that will be sent to the SPF. To mitigate this uncertainty, DOE has developed a sampling plan to: (1) provide characterization of the waste streams input to the saltstone feed tank, (2) ensure that the grout recipe can be modified as needed to compensate for changes in the salt, organic, or solids content of the salt solutions, (3) verify that the waste is nonhazardous, and (4) verify compliance with the WAC (Ketusky, 2005). Although the frequency of sampling is provided in the plan, it is not clear that the sampling frequency for various types of samples is consistent with the schedule for batch waste preparation operations. DOE indicates that process knowledge will be used for chemical and radiological characterization of the transfers to Tank 50, and that material balance may then be used to estimate the chemical and radiological composition of the material transferred to the SPF. Tank 50 contents would be re-baselined with data from samples analyzed quarterly for chemical characteristics and semi-annually for radiological characteristics (Ketusky, 2005).

Process knowledge can be used in the development of inventory estimates; however, past experience with tank farm characterization has demonstrated that process knowledge can result in significant deviations from actual waste stream composition as determined by direct sampling. As indicated by DOE, the uncertainty in Cs-137 activity sent to the SPF may be as large as approximately 70% of the current best estimate, and Cs-137 is one of the most well-characterized radionuclides in the tank farm system. The sampling plan needs to consider the uncertainty in the methods and data used to estimate the radiological composition of the waste sent to the SPF. If a factor of safety was applied to the inventory limits generated with the PA, then the likelihood that material that exceeds the WAC would be sent to the SPF (because of indirect methods of estimating composition) could be reduced. However, DOE has not indicated that it intends to apply any additional measures to account for the uncertainty in the waste characterization. The least uncertain method is to perform adequate sampling of each batch of waste in the feed tank prior to transferring the waste to the SPF to ensure that the WAC can be met. Sampling of each batch of waste in the feed tank prior to transfer to the SPF is needed unless it can be demonstrated through a predictive exercise that indirect methods can adequately predict the concentrations of radionuclides in the saltstone feed tank.

Use of an adequate sampling plan combined with a sum-of-fractions approach to compare the disposal of the actual saltstone inventory to the performance objectives is an acceptable approach to managing uncertainties in the estimation of the radiological source term for the PA.

4.2.4 Infiltration and Erosion Control

The SDF closure concept is illustrated in Figures 5 and 6. After each vault is filled with saltstone, interim closure will involve pouring a clean grout layer between the saltstone and overlying concrete roof. Final closure of the disposal facility will be accomplished by constructing a thick, multi-layer engineered cap, a drainage system, and performing re-vegetation of the site. The drainage system will consist 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 re-vegetated with bamboo that is expected to quickly establish a dense ground cover and limit the growth of pine trees, the most deeply rooted naturally occurring plant type at the Savannah River Site. Through evapotranspiration, the bamboo is expected to remove a large amount of moisture from the soil and decrease infiltration into the underlying disposal system.

The engineered cap serves two primary purposes: to limit infiltration to the waste and to limit erosion. Both are discussed in detail below. By limiting water contact with the waste, the release of radioactivity from advection can be limited. The erosion control barrier will be located approximately one meter below the top of the final closure cap. Approximately 3.2 m (10.5 ft) of material will be between the erosion control barrier and the top of the waste. Most agricultural or resident intruder scenarios consider a nominal excavation depth of 3 m (9.8 ft), therefore proper design, implementation, and function of the erosion control barrier would essentially eliminate direct contact with the waste for intruder receptors, reducing the exposure pathways to primarily that of direct radiation exposure. The remaining material (upon erosion down to the erosion control barrier) will also provide shielding from gamma radiation for intruder receptors. The top layer of the cap will be topsoil and backfill on which bamboo will be grown. Bamboo provides an effective method to reduce infiltration through evapotranspiration. Below the erosion barrier will be an upper drainage layer and a geosynthetic clay layer (GCL), which are designed to limit infiltration.

4.2.4.1 Infiltration

DOE recognizes that limiting the amount of water flowing through the saltstone vaults and contacting the saltstone is important for limiting releases of radionuclides from the vault into the accessible environment. DOE proposes to reduce water fluxes that contact the waste using a multiple barrier system, including: (1) vegetative reduction of infiltration using a dense ground cover for a species with large evapotranspiration demand during institutional control, (2) a closure cap designed to shed water from the vaults using redundant drainage layers, and (3) favorable hydraulic characteristics of the vault and wasteform that limit both advective transport and molecular diffusion within water. The closure cap consists of three functional components separated by backfill, including: (1) an erosion barrier, (2) an upper drainage unit intended to shed water far from the vault, and (3) a lower drainage unit intended to route water around the top and sides of the vault into a buried drainage system.

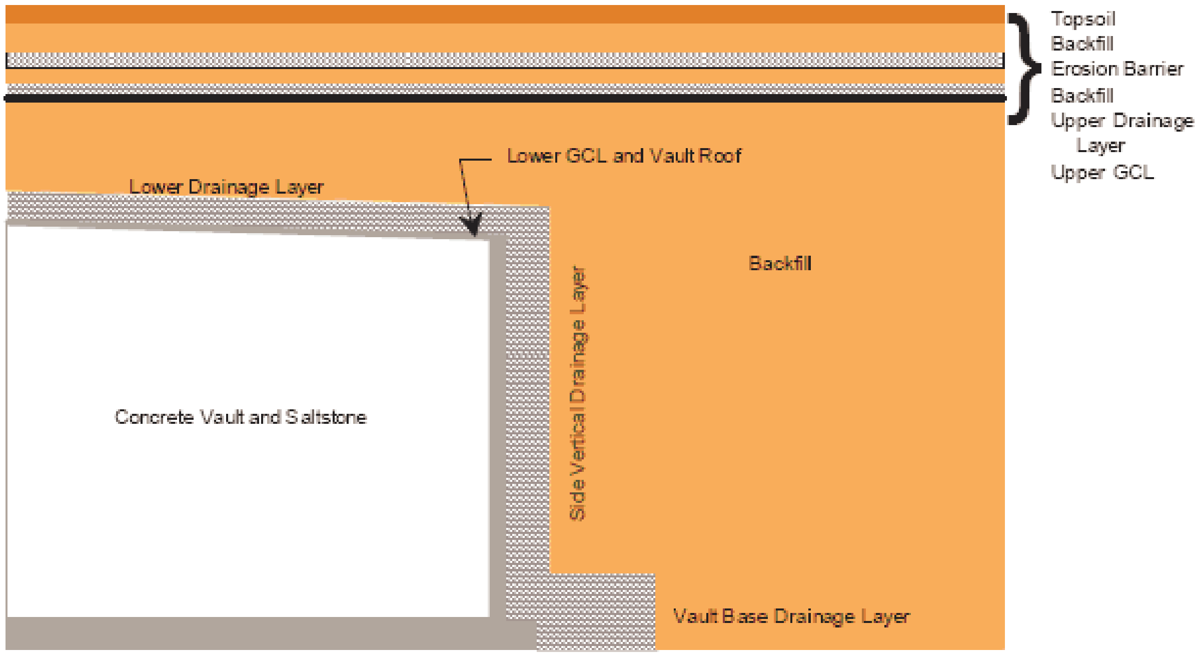

Figure 5 Engineered Closure Cap Side View

Each of these barriers is assumed to degrade over time. Institutional control of the initial ground cover (bamboo) is assumed to be maintained for 100 years. Once institutional control ceases, ground cover is assumed to revert to native pine forest after 200 years. Reversion to pine forest would decrease evapotranspiration, causing an increase in net infiltration. Closure-cap degradation is defined as altering the thickness or hydraulic properties of each of the cap layers over time. Three degradation processes were considered: (1) erosion, which removes topsoil and upper backfill; (2) pine forest root penetration, which punctures the upper GCL, and (3) colloidal clay migration, which moves colloidal clay from backfill into the pores of the underlying drainage layers. Saturated hydraulic conductivity was measured for the porous medium used for the drainage layers, and the manufacturer provided a value for the GCL. Changes in hydraulic properties were inferred from the degradation processes.

4.2.4.2 Erosion Control

The upper layers of the engineered cap will primarily serve as erosion barriers and will consist of a 1 m (3 ft) thick soil cover and a 0.3 m (12 in) rock layer on the top slopes of the disposal cell. The rock erosion barrier will be placed on top of the middle backfill and geotextile fabric to form a barrier to erosion if a gully were to form in the upper soil layer. However, the upper soil layer is designed to be stable during the occurrence of very rare flood events, such that a considerable margin of safety exists for the prevention of erosion. For the side slopes, erosion protection will be provided by a 0.6 m (24 in) rock layer placed directly on top of the slopes.

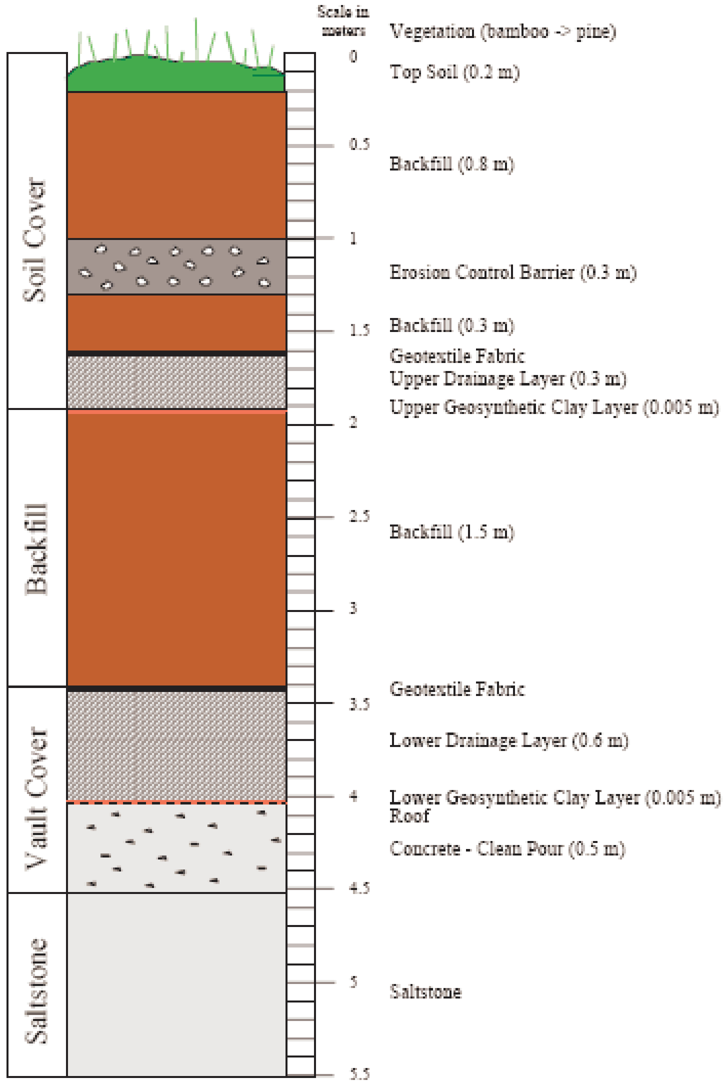

Figure 6 Engineered Closure Cap Layers

The computation of peak flood discharges for various site design features was performed by DOE in several steps. These steps included: (1) selection of a design rainfall event; (2) determination of infiltration losses; (3) determination of times of concentration; (4) determination of appropriate rainfall distributions, corresponding to the computed times of concentration; and (5) calculation of flood discharge. Input parameters were derived from each of these steps and were then used to determine soil cover slopes and rock sizes for erosion protection.

One of the phenomena most likely to affect long-term stability is surface water erosion. DOE used a Probable Maximum Precipitation (PMP) computed by deterministic methods, rather than statistical methods, and the PMP was based on site-specific hydrometeorological characteristics. The PMP has been defined as the most severe reasonably possible rainfall event that could occur as a result of a combination of the most severe meteorological conditions occurring over a watershed. PMP values were estimated by DOE using Hydrometeorological Report No. 51 (HMR-51) (COE, 1978) and HMR-52 (COE, 1982). These reports also provide information on distributing the rainfall that falls over a particular drainage area. A 1-hour PMP of 48.8 cm (19.2 in) and a 5-minute PMP of 15.7 cm (6.2 in) were used by DOE as a basis for estimating a Probable Maximum Flood (PMF) for the smaller areas at the site such as the top and side slopes.

The determination of the peak runoff rate is dependent on the amount of precipitation that infiltrates into the ground during its occurrence. If the ground is saturated from previous rains, very little of the rainfall will infiltrate and most of it will become surface runoff. The loss rate is highly variable, depending on the vegetation and soil characteristics of the watershed. Typically, all runoff models incorporate a variable runoff coefficient or variable runoff rates. Commonly-used models such as the U.S. Bureau of Reclamation (USBR) Rational Formula (USBR, 1977) incorporate a runoff coefficient (C); a C value of 1 represents 100% runoff and no infiltration. Other models such as the U.S. Army Corps of Engineers Flood Hydrograph Package HEC-1 (COE, 1988) separately compute infiltration losses within a certain period of time to arrive at a runoff amount during that time period. In computing the peak flow rate for the small drainage areas at the site, DOE used the Rational Formula (USBR, 1977). In this formula, the runoff coefficient was assumed to be 0.8; that is, DOE assumed that very little infiltration would occur.

The time of concentration is the amount of time required for runoff to reach the outlet of a drainage basin from the most remote point in that basin. The peak runoff for a given drainage basin is inversely proportional to the time of concentration. If the time of concentration is assumed to be smaller, the peak discharge will be larger. Times of concentration and/or lag times are typically computed using empirical relationships. Times of concentration for the riprap design were estimated by DOE using the Kirpich Method (USBR, 1977). This method is generally accepted in engineering practice.

After the PMP is determined, it is necessary to determine the rainfall intensities corresponding to shorter rainfall durations and times of concentration. A typical PMP value is derived for periods of about one hour. If the time of concentration is less than one hour, it is necessary to extrapolate the data presented in the various hydrometeorological reports to shorter time periods. To accomplish this, DOE used a procedure recommended in HMR-52. This procedure involves the determination of rainfall amounts as a percentage of the one-hour PMP, and computes rainfall amounts and intensities for very short periods of time. To calculate peak flood flows, the 5-minute PMP rainfall intensity was determined to be about one-third of the 1-Hr PMP, or about 15.7 cm (6.2 in).

To estimate PMF peak discharges for the top and side slopes, DOE used the Rational Method (Chow, 1959). This method is a simple procedure for estimating flood discharges and is recommended in NUREG-1623 (Johnson, 2002). In using the Rational Method, DOE assumed a runoff coefficient equal to 0.8. DOE also assumed a flow concentration factor of about 5 and a factor of safety of about 1.35 that would be used to increase the flows for design purposes. For a maximum top slope length of 140 m (450 ft) (with a slope of 0.015) and a side slope length of 18 m (60 ft) (with a slope of 0.33), DOE estimated the peak flow rates to be about 0.29 m^3/s-m width (3.1 cubic feet per second per foot of width [cfs/ft]) for the top slope and 0.33 m^3/s-m width (3.5 cfs/ft) for the side slope. PMF flow rates for overland flow for the downstream apron were estimated by DOE and are slightly less than the flow rates for the side slopes.

The ability of a riprap layer to resist the velocities and shear forces associated with surface flows over the layer is related to the size and weight of the stones which make up the layer. Typically, riprap layers consist of a mass of well-graded rocks which vary in size. Because of the variation in rock sizes, design criteria are generally expressed in terms of the median stone size, D_{50}, where the numerical subscript denotes the percentage of the graded material that contains stones of less weight. For example, a rock layer with a minimum D_{50} of 10 cm (4 in) could contain rocks ranging in size from 1.9 cm (0.75 in) to 15 cm (6 in); however, at least 50% of the weight of the layer will be provided by rocks that are 10 cm (4 in) or larger.

The top portion of the cell will be protected by a 1 m (3 ft) thick soil layer. DOE proposes that the soil cover will be stable and will provide the necessary erosion protection for a 10,000-year period. The top slope was evaluated for erosional stability by DOE using the permissible velocity procedure discussed in NUREG-1623. Using a peak discharge of about 0.29 m^3/s-m (3.1 cfs/ft), the maximum velocity was computed to be about 0.79 m/s (2.6 ft/s), less than the maximum permissible velocity of 0.91 m/s (3.0 ft/s). However, to increase the margin of safety, in the unlikely event that a gully would occur in the soil cover for any reason, DOE proposes to provide a riprap layer under the 1 m (3 ft) soil cover to prevent further vertical erosion into the geotextile and drainage layers. This size of this riprap was computed assuming that a flow concentration factor of 5 would occur, and a factor of safety of 1.35 would be applied. Using the design procedure recommended in NUREG-1623, DOE computed the required rock size to resist further gullying to be about 6.6 cm (2.6 in). DOE proposes to place a 0.3 m (12 in) layer of riprap with a D_{50} of about 10 cm (4 in) under the soil cover. For the side slopes of the cell, DOE proposes to use a 0.6 m (24 in) layer of rock with an average D_{50} of 0.25 m (10 in). Methods discussed in NUREG-1623 were used to determine a required rock size of 20 cm (7.7 in).

The design of the apron for the cell must be adequate to withstand forces from several different phenomena and is based on the following general concepts: (1) provide riprap of adequate size to be stable against overland (downslope) flows produced by the design storm (PMP), with allowances for turbulence along the downstream portion of the toe; (2) provide uniform and/or gentle grades along the apron and the adjacent ground surface such that runoff is distributed uniformly onto natural ground at a relatively low velocity, minimizing the potential for flow concentration and erosion; (3) provide an adequate apron length and quantity of rock to allow the rock apron to collapse into a stable configuration if gullying occurs and erodes toward the site, and (4) provide an apron with adequate rock size to resist flows that will occur laterally along the apron. Using design criteria suggested in NUREG-1623, DOE determined that riprap with a D_{50} of 0.25 m (10 in), a layer thickness of 0.91 m (36 in), and a width of 6 m (20 ft) would be placed in the apron area.

Depending on the rock source, variations occur in the sizes of rock available for production and placement, and it is therefore necessary to ensure that these variations in rock sizes are not extreme. Examples of acceptable gradations are provided in NUREG-1623. The estimated average D_{50} values to be used as the basis for the design of well-graded mixture of rock to resist the shear forces of the PMF peak discharge were determined to be 10 cm (4 in) for the top slopes and 25 cm (10 in) for the side slopes and apron. Riprap gradations and layer thicknesses were developed by DOE using criteria suggested in NUREG-1623.

Rock durability is defined as the ability of a material to withstand the forces of weathering. Factors that affect rock durability are: (1) chemical reactions with water, (2) saturation time, (3) temperature of the water, (4) scour by sediments, (5) windblown scour, (6) wetting and drying, and (7) freezing and thawing. To assure that the rock used for erosion protection remains effective for long periods of time, potential rock sources must be tested and evaluated. A procedure for determining the acceptability of a rock source is presented in NUREG-1623. In general, rock durability testing is performed using standard test procedures, such as those developed by the American Society for Testing and Materials (ASTM). The ASTM publishes and updates an Annual Book of ASTM Standards (ASTM, 1995), and rock durability testing is usually performed using these standardized test methods.

In accordance with the criteria outlined in NUREG-1623, DOE provided information regarding testing, inspection, and quality control procedures to be used for the erosion protection materials. The information included programs for durability testing, gradation testing, and rock placement. DOE provided information regarding the testing program that will be used to document the durability of the rock source selected. Using the scoring procedure discussed in NUREG-1623, DOE intends to test the rock and to use only rock that achieves a minimum score of 80. DOE's proposed rock durability testing program includes the following tests: (1) bulk specific gravity, (2) absorption, (3) sodium sulfate soundness, (4) L.A. abrasion at 100 cycles, and (5) Schmidt hardness.

DOE proposes that rock gradation testing will be performed in accordance with standard ASTM test procedures, as suggested in NUREG-1623. DOE provided a placement program where riprap will be placed in accordance with ASTM standard test procedures and procedures suggested in NUREG-1623. Further, because the rock layer will be covered with soil, it is important to assure that the rock voids are uniformly filled and that differential settlements are minimized. To accomplish this, DOE intends to fill the rock voids with a grout mixture (flowable fill). Placement of this type of material will greatly enhance the stability of the rock layer and adds a considerable margin of safety to the overall design.

4.2.5 NRC Evaluation – Infiltration and Erosion Control

It is important to note that design for long-term infiltration control and design for long-term erosion control have different objectives and different degradation mechanisms. Acceptability of a design for one does not necessarily mean that an acceptable design has been achieved for the other.

4.2.5.1 NRC Evaluation - Infiltration

The DOE approach to assessing the long-term performance of the engineered cap to limiting infiltration is based on a reasonable conceptual model. DOE considered three degradation processes: (1) erosion, which removes topsoil and upper backfill; (2) pine forest root penetration, which punctures the upper GCL; and (3) colloidal clay migration, which moves colloidal clay from the backfill into the pores of the underlying drainage layers. The approach of using the HELP model with assumptions about pine tree root penetrations over time is a reasonable conceptual model for estimating deterioration of the upper GCL over time. The amount of infiltration and the timing of infiltration through the upper GCL results in long-term near surface infiltration rates that are not significantly lower than natural recharge rates may be expected to be for the site.

NRC staff requested that DOE provide its estimate of the amount of infiltration available to the wasteform that may be diverted due to the low permeability vault and saltstone. This type of information would provide an indication of how much credit was being taken by DOE for long-term cap performance. DOE provided a figure of fractional flow through the saltstone as a function of time for the base case PA (WSRC, 2005a). The fractional flow reaches a maximum of approximately 2×10^{-4} at 10,000 years. It is not clear how effective the lower drainage layer of the engineered cap would be at limiting infiltration to the waste at higher infiltration rates through the cap or for larger amounts of degradation to the saltstone and vaults

Independent analysis by the NRC staff suggests that several factors mediate the closure cap performance in limiting fracture flows. The upper GCL is the first key engineered barrier, and will fail as roots penetrate the layer. The lower drainage layer is the second key barrier, and could fail as it is plugged with colloidal clay. A lower GCL is placed on the roof of the vaults; however, DOE has taken no performance credit for the lower GCL in the analysis. The lower GCL mediates flow rates and could control whether or not a fracture is filled with a porous medium, both largely depending on whether the GCL ruptures during settlement. The DOE analysis suggests that it is credible to consider the scenario where the GCL allows the fracture to be filled with the overlying porous medium, while exposure to the environment degrades the saltstone near the fracture. This scenario would be credible for every fracture that stretches the GCL beyond its limits, either by opening at the top or by vertical displacement. Since the fracture would be filled with the same material as the drainage layer, there would be no capillary effect precluding flow into the fracture, although flow rates might be small. There is no known relevant experience for the long-term performance of GCLs.

An independent NRC analysis suggests that the model used by DOE to estimate the effects of colloidal clay plugging the lower drainage layer may not be conservative. If the hydraulic properties of the lower drainage unit are modified uniformly throughout the layer to reflect partial plugging by colloidal clay, open-fracture flow may be allowed in less than a third of the time predicted by the DOE model, in which complete plugging occurs in a discrete sublayer within the lower drainage unit. The NRC staff expects that colloidal clay transport will result in the highest degree of clay deposition occurring at the top of the lower drainage layer, with clay deposition that may be smeared from the sharp front implied by the DOE model. The NRC staff recognizes that the rates of colloidal clay transport used by DOE are based on literature values (Phifer and Nelson, 2003), while the soils at the SRS have low pH values that may modify colloidal transport. Because the integrity of the lower drainage layer behavior affects every fracture, uncertainty in drainage layer behavior is expected to be important to performance. Uncertainty in the estimated rate of degradation of the lower drainage layer could be reduced through further studies or could be mitigated through additional conservatism in cap design.

The DOE model for lateral diversion in the lower drainage layer, as implemented, may exaggerate lateral diversion in the PA exercises due to the calculational approach. The approach represents the degraded lower drainage layer as an equivalent anisotropic medium in the PORFLOW simulations. The horizontal and vertical hydraulic conductivities of the lower drainage layer are calculated assuming flow is in series and in parallel, respectively, which NRC staff agrees is a reasonable approach. However, it is only appropriate to perform this calculation using saturated hydraulic conductivities when the medium is saturated. For example, the ratio of horizontal-to-vertical conductivity using the DOE approach increases from 1.1 to 233 over the period of simulation. When using representative suction heads of 100, 1,000, and 5,000 cm (3.3, 33, and 164 ft) in both media simultaneously, the maximum ratio of horizontal- to-vertical unsaturated conductivity is 280, 3.2, and 1.2, respectively. This discrepancy will presumably reduce predicted doses, since the hydraulic conductivities of the vault are small, and it may not have a large effect unless fracture flows are initiated.

Aside from the shortcomings noted above, NRC staff believes the DOE approach to assessing long-term cap performance is reasonable. However, DOE provided limited model support to justify the numerical modeling results for long-term infiltration through the engineered cap. Recent experience with field-scale observations of engineered caps at a number of different types of sites suggest that, in many cases, performance may be significantly less than predicted by numerical modeling (Albright, et al., 2004). Model support is needed to ensure that the idealized and simplified numerical calculations are appropriately representing the real-world system. The rate at which the lower drainage layer plugs is determined by the rate at which colloidal clay migrates into the layer. With the current modeling approach, the likelihood of flow in fractures (if present) is determined by the saturation of the lower drainage layer, which in turn is primarily driven by the hydrologic properties of the lower drainage layer. As shown in Section 4.2.16, both the amount of infiltration to the waste and the likelihood of fracture flow can significantly influence whether the disposal system can meet the performance objective of protection of the public. The processes being modeled are highly uncertain, and adequate justification is needed that the modeling has appropriately accounted for uncertainties or is sufficiently conservative based on the model support.

4.2.5.2 NRC Evaluation - Erosion Control

The primary purpose of the erosion control system is to ensure that a thick cover of soil is maintained over the waste to eliminate exposure pathways to hypothetical inadvertent intruders. To mitigate the potential effects of surface water erosion, the NRC staff considers that it is very important to select an appropriately conservative rainfall event on which to base the flood protection designs. Further, the staff considers that the selection of a design flood event should not be based on the extrapolation of limited historical flood data, due to the unknown level of accuracy associated with such an extrapolation. No recurrence interval is normally assigned to the PMP; however, the staff has concluded that the probability of such an event being equaled or exceeded is very low. Accordingly, the PMP is considered by the NRC staff to provide an reasonable design basis. DOE used HMR-51 and HMR-52 to estimate PMP amounts and staff review of these procedures indicate that the PMP amounts are reasonable for the small drainage areas at the site.

Based on a review of the calculations, including the time of concentration, rainfall intensity, and runoff, the staff concludes that the estimates for peak flood discharges are conservative and are, therefore, reasonable. In particular, for the design calculations DOE used a runoff coefficient of 0.8. Based on the conservatism associated with the use of a 5-minute PMP of 15.7 cm (6.2 in) and a resulting rainfall intensity of 190 cm/hr (74 in/hr), the staff concludes that this is a reasonable assumption. Times of concentration for the riprap design were estimated by DOE using the Kirpich Method (USBR, 1977). This method is generally accepted in engineering practice and is considered by the staff to be appropriate for estimating times of concentration at this site. Based a review of the calculations provided, the staff concludes that the time of concentration values used by DOE were appropriately derived. To calculate peak flood flows, the 5-minute PMP rainfall intensity was determined to be about one-third of the 1-hr PMP, or about 15.7 cm (6.2 in). Based on a review of this aspect of the flooding determination, the staff concludes that the computed peak rainfall intensities are reasonable.

Staff reviewed the DOE calculations for stability of the top slope, side slopes, and apron. Based on review of the calculations, the conservatism associated with the design discharge, and the ability of the cover to provide adequate protection against the computed velocities and shear stresses, the staff concludes that the top slope should be stable under extreme flooding conditions. The proposed rock size for the side slope is adequate, and the apron design is acceptable (based on the conservatism associated with the computation of flow rates). Based on the acceptability of using procedures suggested in NUREG-1623, the riprap design should achieve the objectives. In addition, the proposed information and procedures are sufficient to ensure acceptable placement of the riprap and fill into the riprap voids.

DOE's proposed gradations are adequate to assure acceptable protection. Based on review of the durability test procedures and ASTM gradation test procedures, the staff concludes that the programs are appropriate. The gradation testing program will ensure that rock layers with acceptable gradations are provided.

Based on review of the information submitted by DOE and on independent calculations, the NRC staff concludes that DOE has provided sufficient information to conclude that the erosion protection design is adequate to provide reasonable assurance of long-term stability of the closure cap for erosion control purposes. It should be emphasized that DOE has provided a conceptual design, and this design could change in the future, depending on the number and location of vaults. Staff's conclusions on this conceptual design is based largely on the conservatism associated with the design, as proposed. There may be several individual design parameters which may not be acceptable if some of the other conservative design parameters are changed or removed. When making design changes, DOE should ensure appropriate conservatism is maintained.

4.2.6 Wasteform and Concrete Vault Degradation

DOE evaluated several alternatives for the final disposal of decontaminated salt waste and the preferred method it selected was the conversion of the waste to a cementitious form suitable for shallow land disposal on the SRS. A cementitious wasteform was selected because it is: (1) resistant to leaching; (2) tolerant to variations in waste composition, waste loading, and radiation levels; (3) produced from low cost and readily available inorganic materials; and (4) suited to high volume production and emplacement using known concrete production technology (Cook and Fowler, 1992). The bases for this decision were discussed in the Environmental Impact Statement for Defense Waste Processing Facilities at the Savannah River Site (DOE, 1982). Subsequent to this decision, DOE undertook extensive studies to optimize the production process and the solid waste product and to select an appropriate landfill design and location. The final wasteform evolved from a mixture of salt solution and cement to its present formulation containing a blend of slag, fly ash, and cement or lime mixed with the salt solution. The disposal technology has evolved from a series of simple trenches that would be filled with waste and covered with native soil to the current design of above-grade concrete vaults that provide an engineered secondary containment barrier for the waste. The following description is taken mostly from Cook and Fowler (1992) and Cook, et al. (2005).

The current wasteform, referred to as saltstone, is a solid that is the product of chemical reactions between a salt solution and a blend of cementitious materials (blast furnace slag, fly ash, and a lime source). The hydration of slag in the saltstone releases sulfide species, predominantly $S^{2!}$, into the pore fluid, which imposes a strongly reducing redox potential on the system and chemically binds several contaminants as insoluble species. Thus, the propensity of these contaminants to leach from the solid wasteform is reduced. A broad range of wasteform precursor compositions has been shown via leach testing to yield saltstone that qualifies as nonhazardous waste, as defined by EPA. The recommended composition for saltstone, using a nominal salt solution blend, is 47 wt% salt solution (29 wt% salt), 25 wt % of Grade 120 slag, 25 wt% of Class F fly ash, and 3 wt% of Type II Portland cement or lime.

Chemical reactions between the components in the salt solution and the dry blend changes the composition of the saltstone compared to the starting materials. Cr (VI) species, technetium (VII) species, and salt solution contaminants that form sparingly soluble sulfides (Hg, Co, Ni, Zn, Tc, Ru, Rh, Sb, Sn) react with components of the slag to form insoluble phases that are not readily leached from the saltstone wasteform. Water and soluble Al (III), Ca (II), Ba (II), and Sr (II) species are incorporated into the cement matrix as the dry materials hydrate and the saltstone sets. Tc (VII) is believed to react with components of the slag to form Tc_2S_7, whereas Cr (VI) in the salt solution is believed to be reduced to a lower oxidation state and precipitated as a Cr (III) hydroxide solid. Strontium and barium are incorporated as aluminosilicates within the aluminosilicate structure of the cement paste. These less soluble forms effectively fix the contaminants, thus reducing their potential release through groundwater pathways.

When first prepared, the saltstone grout has the consistency and flow characteristics of latex paint and is readily pumped from the SPF to a cell in a disposal vault. The disposal vaults serve as a form to support the saltstone grout until it sets into a solid, monolithic wasteform. The saltstone itself provides primary containment of the waste, but the walls, floors, and roof of the vaults provide a secondary containment barrier for the contaminants in the waste and isolate the waste from direct contact with the environment.

In the proposed disposal site layout, up to 14 additional vaults made of reinforced concrete will be constructed. These vaults will each have dimensions of approximately 60 m (197 ft) wide by 180 m (591 ft) long by 7.6 m (25 ft) high, divided into twelve cells approximately 30 m (98 ft) wide by 30 m (98 ft) long by 7.6 m (25 ft) high. One vault that has already been constructed and filled with material, designated as Vault 1 on the site layout plan, is approximately 30 m (98 ft) wide by 180 m (591 ft) long by 7.6 m (25 ft) high, divided into six cells with the same cell dimensions as those in the larger vaults. Operationally, the cell of the vault will be filled to a height of approximately 7.5 m (25 ft) with saltstone, and then a layer of uncontaminated grout will be poured to fill in the space between the saltstone and the sloped roof. Vaults 1 and 4 have already been built at the SDF. These two vaults are made of reinforced concrete containing slag. Because the current saltstone formulation also includes slag, the use of slag in the construction of future vaults to mitigate the release of certain contaminants is redundant and may be reduced or eliminated to reduce costs and to improve the resistance to cracking due to shrinkage during curing.

As mentioned previously, the saltstone wasteform provides primary containment of the contaminants, and the concrete vault serves as a secondary containment barrier mitigating the release of contaminants to the environment. The performance of these barriers is important in order to meet the NRC performance objectives in 10 CFR Part 61, Subpart C. Thus, the methods used to analyze the long-term performance of the SDF must account for potential mechanisms of contaminant release from the facility, and the potential mechanisms for loss of integrity of the SDF engineered barriers. At a conceptual level, mechanisms that lead to the release of contaminants from the saltstone consist of dissolution, diffusion, dispersion, and advection. The rates of contaminant release by these mechanisms are controlled principally by (1) the moisture flux to the saltstone through the vault from the overlying soil, (2) the solubility and the diffusive and sorption properties of the contaminants, and (3) the chemical properties (e.g., redox condition) and physical characteristics (e.g., hydraulic conductivity, presence, and density of fractures) of the wasteform and vault. For example, the DOE analysis relies on the presence of blast furnace slag in the saltstone to produce chemical conditions that result in low solubilities and increased retardation for certain contaminants. The hydraulic conductivity of the vault and saltstone will affect the rate at which contaminants can migrate to the surrounding soil. Degradation of the vaults and monoliths through chemical (e.g., sulfate attack, carbonation, leaching, and rebar corrosion) or physical (e.g., cracking due to settlement of foundation soil, seismic activity) processes is likely to increase contaminant release due to increased advection of contaminants.

In conducting its PA of the facility, DOE considered the various mechanisms of release to estimate the source term and release of contaminants. Both diffusive and advective transport processes were addressed. To model contaminant transport in the near field, there was a need to estimate the contaminant concentrations in the pore fluid based on the concentrations in the saltstone. However, relating inventory in the saltstone to pore fluid concentration is complicated by various processes, such as precipitation/dissolution reactions, aqueous complex formation, and sorption. DOE acknowledged that these processes are poorly understood and difficult to quantify for the SDF. Thus, concentrations in the saltstone pore fluid were derived using models that considered only reversible linear sorption (i.e., distribution coefficients [K_d values]). In its 1992 PA (Cook and Fowler, 1992), DOE used geochemical analyses to derive the initial pore solution concentrations of Tc-99, I-129, and nitrate, but applied K_d values for saltstone constituents considered by DOE to be relatively unimportant with respect to acceptable performance of the facility. In its 2005 special analysis (Cook, et al., 2005), DOE used K_d values taken from a report by Bradbury and Sarott (1995), which was considered to provide the best conservative K_d values for cement-water systems available in literature. The K_d values selected from the report of Bradbury and Sarott (1995) are values that apply to reducing environments. In selecting those values, DOE assumed that the rate of oxidation of the saltstone was sufficiently slow such that the saltstone remained reducing throughout the 10,000 year simulated performance due to the presence of slag.

Degradation that alters the integrity and permeability of the saltstone monoliths and vaults also was considered. Thus, scenarios with an intact vault and a degraded vault were evaluated. However, DOE recognized that “the timing and extent of degradation are not readily predictable due to enormous uncertainties in conditions for thousands of years” (Cook and Fowler, 1992). Thus, in the 1992 PA (Cook and Fowler, 1992), cracking of the vaults and monolith was chosen to represent the increased permeability of the waste and the vaults. For simplicity, cracking was represented by vertical cracks that fully penetrate the vault and saltstone, with a spacing of 3 m (10 ft) and an average aperture of 0.005 m (0.016 ft). The 1992 PA also included sensitivity and uncertainty analyses on various processes and related parameters to determine their significance to facility performance. The parameters to which the results were most sensitive were the hydraulic conductivity and diffusivity of concrete and saltstone. For degraded vaults, the results were most sensitive to the depth of perched water, crack spacing, and the K_d values.

In the 2005 special analysis (Cook, et al., 2005), cracks in the saltstone were not included in the analysis on the basis that cracks will open either at the top or at the bottom and will be pinched closed at the opposite end, preventing the development of through-wall cracks (Peregoy, 2003). Instead, degradation of the closure system, the saltstone wasteform, and the vault, regardless of the mechanism, was represented by using material properties that varied for each time interval. For example, it was assumed that the hydraulic conductivities of saltstone and concrete would increase as time proceeds, which would gradually increase water percolation through the vault. The hydraulic conductivities of saltstone and concrete in the time interval of 0 to 100 years were set to 1 x 10^{-11} and 1 x 10^{-12} cm/s (1 x 10^{-5} and 1 x 10^{-6} ft/yr), respectively, and were increased to 1 x 10^{-9} cm/s (1 x 10^{-3} ft/yr) over a 10,000-year period through eight steady-state stages. The degradation rate for concrete was assumed to be faster than for saltstone because the concrete will be exposed to the environment and would be more vulnerable to attack by sulfate, chloride, and other chemical reactions. DOE acknowledged that the assumptions regarding the changes in hydraulic properties over time are based on professional judgment because actual data over the time periods of interest do not exist.

4.2.7 NRC Evaluation – Wasteform and Concrete Vault Degradation

The hydraulic conductivities of nondegraded saltstone and concrete vault used in the PA were based on limited data. The hydraulic conductivity of the nondegraded concrete vault (1.0 × 10^{-12} cm/s [1.0 × 10^{-6} ft/yr]) used in the analyses was based on measurements by Core Laboratories, Inc. of samples from the Savannah River Site E-area low-level waste disposal vault (WSRC, 1998). The hydraulic conductivity of nondegraded saltstone (1.0 × 10^{-11} cm/s [1.0 × 10^{-5} ft/yr]) was based on measurements on saltstone grout specimens by Core Laboratories, Inc. (Yu, et al., 1993). For comparison purposes, values of hydraulic conductivities of concrete and cement pastes available in the literature were tabulated by DOE (WSRC, 2005a). The tabulated values indicate that 1.0 × 10^{-12} cm/s (1.0 × 10^{-6} ft/yr) for nondegraded vault is on the low end of the range of literature values and is similar in magnitude to that of high performance concrete that uses a super plasticizer. Higher values (1.1 × 10^{-10} to 2.3 × 10^{-9} cm/s [1.1 × 10^{-4} to 2.4 × 10^{-3} ft/yr]) of hydraulic conductivity were measured for samples taken from the Savannah River Site Z-area saltstone vault. DOE explained that the higher hydraulic conductivity for the Z-area vault samples was due to the lesser quality of workmanship in the construction of the Z-area vault compared to the E-area vault. Data from (Malek et al., 1985) for earlier formulations of saltstone provided hydraulic conductivity values that ranged from 2.5 × 10^{-6} cm/s [2.5 ft/yr] to <1.0 × 10^{-11} cm/s [1.0 × 10^{-5} ft/yr] for nondegraded saltstone. Measurements were presented at different curing times. It is not clear if the variability in the measurements is a function of curing time or if the variability results from the measurement technique. The data suggest that longer curing times may result in higher permeabilities, although it cannot be determined whether the result is an artifact of the uncertain measurement technique. Because the hydraulic conductivity data being relied upon by DOE represents limited samples over a small range of curing times, DOE should consider performing additional tests for the current saltstone formulation.

DOE indicated a range of saltstone compositions over which acceptable saltstone can be produced based on results of Toxicity Characteristic Leaching Procedure (TCLP) tests (WSRC, 2005b). In the RAI, NRC requested justification that the physical properties of saltstone would not vary significantly over the broad range of composition for which the TCLP tests were performed. DOE indicated that the nominal composition of saltstone has not varied significantly on a daily or batch basis (WSRC, 2005b). The larger range of compositions were intended to accommodate the State-approved operating permit for the facility. Because there was a fairly significant amount of variability in the TCLP test results, significant deviation from the nominal saltstone composition may result in a saltstone wasteform composition for which tests have not been performed to develop hydraulic conductivity and effective diffusivity values.

Because of the potentially problematic interactions of Tank 48 waste with other tank farm waste or with the proposed treatment processes, DOE has indicated Tank 48 waste will be sent to SPF without treatment; that is, there will be no removal of highly radioactive radionuclides from the Tank 48 waste before it is solidified. DOE conducted TCLP tests on samples containing Tank 48 waste and the results were found to be acceptable. However, the technical basis (e.g., hydraulic conductivity and effective diffusivity) for the performance of a saltstone wasteform containing Tank 48 waste has not been developed.

In the 2005 special analysis (Cook, et al., 2005), the degradation of the concrete vault and saltstone was modeled by increasing the hydraulic conductivity of the vault and saltstone to 1.0×10^{-9} cm/s (1.0×10^{-3} ft/yr) at 10,000 years. This value was selected based on professional opinion of DOE staff. The professional opinion is based on: (1) an assessment of potential degradation mechanisms identified in published literature; (2) a simplified assessment of the consequences of these mechanisms; (3) a determination that freeze/thaw and erosion will not occur because the vault and wasteform will remain buried over the period of performance; and (4) the assumption that the effect of environmental factors such as acid rain and leaching by infiltrating water would be significantly less than the effect of cracking (WSRC, 2005a). Recent sensitivity analyses presented by WSRC (2005a) demonstrated that an increase in degradation rate results in an increase in calculated dose. For example, increasing the final hydraulic conductivity to a value typical of a clayey-sandy soil (1.0×10^{-6} cm/s [1.0 ft/yr]), instead of the value used in the 2005 special analysis, increased the calculated dose by a factor of 320 (WSRC, 2005a). The calculated dose (when properly scaled, see Section 4.2.16) would be approximately 1.1 mSv/yr (110 mrem/yr) when assuming a hydraulic conductivity of 1.0×10^{-6} cm/s (1.0 ft/yr), which would exceed the performance objective of 0.25 mSv/yr (25 mrem/yr). Furthermore, an increase in hydraulic conductivity would also adversely affect the reducing capacity of the saltstone due to diffusion or advection of oxygen into the monolith. A combination of increased fluid flow and oxidizing conditions would lead to a calculated dose higher than 110 mrem/yr.

In general, the NRC staff agree with the qualitative assessment of the degradation mechanisms for saltstone. However, given that: (1) the calculated releases from the SDF are sensitive to the values of hydraulic conductivity of the vault and saltstone; and (2) “the timing and extent of degradation are not readily predictable due to enormous uncertainties in conditions for thousands of years” (Cook and Fowler, 1992, Section 3.1.3.5), it would be useful to reduce the uncertainties associated with the hydraulic conductivity and long-term integrity of the vault and saltstone. Additional laboratory measurements of initial hydraulic conductivity, as well as long-term tests or monitoring studies designed to evaluate the long-term durability of the saltstone and concrete vault, would help reduce these uncertainties. The vaults into which the saltstone will be poured contain rebar and fill pipes, in the current design, that would be exposed to the environment. Unless mitigated by future design modifications, consideration should be given to the effects of rebar and fill pipe corrosion on the integrity and projected lifetime of the vaults in future degradation calculations. In the response to action items, DOE indicated that if penetrations in the vault provide an unacceptable moisture flow pathway, the closure plan for the facility will be revised to provide modifications to the design (WSRC, 2005a).

4.2.8 Release and Near-Field Transport

DOE assumed that the vault and saltstone would have several fractures due to differential settlement, with fractures widening gradually over time. Fracture characteristics were based on a structural analysis (Peregoy, 2003) that suggests that the cracks will be preferentially created at joints between concrete pours, and fully penetrate the vault and saltstone, but will only be open on one end (either top or bottom). Vertical offsets between adjacent blocks were not estimated. The cementitious materials of the vault and wasteform were assumed to gradually increase their hydraulic conductivity over time to account for degradation. Hydraulic conductivity measurements were obtained on concrete and saltstone core samples; these measured values are used for early times. Changes over time were based on engineering judgment. An analysis suggesting that the suction heads were so large that fractures would not flow under steady conditions (Cook, et al., 2005, Appendix A.4) was used to justify neglecting fractures in most analyses. In other analyses, fractures were imposed as a series of discrete features with properties similar to gravel.

DOE's groundwater pathways analysis of radionuclide releases used a sequence of three computer codes. The HELP code (EPA, 1994a,b) was used to calculate net infiltration and water balances from the ground surface through the upper geosynthetic clay liner. The PORFLOW code (Analytical and Computational Research, Inc., 2002) used the water fluxes through the upper GCL calculated by HELP as an upper boundary condition and calculated water fluxes in the vadose zone below the upper GCL, within the vault, and in the vadose zone above the saturated zone to calculate a series of steady-state flow fields. In a subsequent step, PORFLOW used the flow fields to calculate the time history of radionuclide releases to the saturated zone. A separate PORFLOW model was used to calculate a steady-state regional water balance. The time history of radionuclides released to the vadose zone was combined with the steady-state saturated zone water fluxes to calculate radionuclide transport in the saturated zone.

Water balances in the vadose zone were calculated as a series of steady-state periods, representing discrete intervals over the analysis. For each period, hydraulic properties were imposed to account for degradation of the closure cap and vault, the HELP model calculated steady-state water fluxes past the upper GCL, and the PORFLOW model calculated steady-state water fluxes below the upper GCL. These water fluxes are held fixed for the duration of a steady-state period while radionuclide release and transport was calculated using PORFLOW. At the end of each period, a new steady-state flow field was created, concentrations were adjusted to account for change in cell saturation, and the transport simulation continued. This procedure can result in non-physical transport responses at the start of a new period that would be smoothed out if the flow fields changed gradually over time.

A number of sensitivity analyses were performed to evaluate the impact of uncertainty in parameters or models. Sections 4.2.15 and 4.2.16 provide a discussion of the DOE analyses and the NRC staff interpretation of those analyses.

4.2.9 NRC Evaluation – Release and Near-Field Transport

DOE is concerned with three primary groundwater-pathway release scenarios: (1) diffusive release via concentration gradients in the pore fluid of the saltstone to the environment outside the vaults; (2) bulk movement of water through the saltstone matrix; and (3) focused release via fracture flow in a few discrete gaps caused by settlement. DOE suggests that fracture flow does not occur until the lower drainage layer fails through plugging resulting in increased saturation that can overcome the capillary barrier effect imposed by the presence of the differing properties of the geologic and engineered materials (Rosenberger, et al., 2005). The DOE conclusion is that the vault system will likely meet performance objectives if the closure cap and vault have properties similar to those used by DOE for its base case. However, there are certain areas in which the DOE modeling assumptions are not strongly supported, and plausible alternative assumptions in these areas may lead to situations that will not meet the performance objectives.

The NRC staff's understanding of the physical processes relevant to water flow and radionuclide releases are summarized in order to put comments on the DOE modeling into context. The following description does not consider releases from disruptive processes (e.g., human intrusion) or gaseous releases through the air pathway. The staff believes that concentrations within the pore water are spatially uniform within most of the saltstone matrix as long as the emplacement stream is unchanged during emplacement. Since the movement of water is slow within the matrix, dissolved radionuclide concentrations should have sufficient time to reach equilibrium with the radionuclides within the saltstone matrix. However, radionuclide concentrations may be significantly altered near the edges of the saltstone matrix due to interactions with the external environment. These interactions include movement of water, oxygen, and radionuclides across the saltstone boundary.

Movement of radionuclides out of the saltstone matrix will occur as diffusion from the edge of the saltstone through the concrete vault into the accessible environment. Advective transport will also occur as water moves across the boundary of the saltstone, carrying dissolved radionuclides with the water. Advective transport tends to counteract diffusive transport where water enters the saltstone and augments diffusive transport where water exits.

Similar effects will regulate the movement of oxygen into the saltstone, potentially creating an oxidized layer that allows certain radionuclides (e.g., Tc-99) to be more mobile where the saltstone meets the accessible environment. Oxygen will diffuse in the gas phase more readily than the liquid phase and may move into the vault more quickly than dissolved radionuclides are released, unless the vault materials are completely liquid-saturated or a gas-phase diffusion barrier is incorporated into the vault design. DOE did not simulate transport of the gas phase; instead, DOE assumed that, outside of the saltstone block, infiltrating water was in equilibrium with atmospheric concentrations of oxygen and then simulated transport of dissolved oxygen in saltstone based on the assumption that the saltstone would remain completely saturated with water. If the saltstone is partially saturated, and oxygen can move through the saltstone in the gas, rather than the liquid phase, it is reasonable to use a coefficient of diffusion for oxygen that is larger than for water-saturated conditions. If the rate-limiting portion of slag capacity consumption is due to transport, the implication is that the oxidized rind may be thicker than estimated by DOE.

In an alternative conceptual model, conditions at the edge of the saltstone matrix may play the dominant role in determining radionuclide releases. When water fluxes are small (as in the DOE base case), diffusion is the primary release mechanism and releases may occur around the entire vault boundary. When water fluxes are large, advection is the primary release mechanism and calculated releases are dominated by advection out of the bottom of the vault. Since essentially all oxidation occurs where the radionuclides contact the external environment, a highly oxidized rind over a few volume percent of the saltstone may govern release rates, which is quite different from the situation where the same total oxidation is uniformly spread throughout the saltstone matrix. Diffusive transport is the primary transport mechanism in the DOE base case. Because diffusive transport is directly proportional to the concentration gradient in the fluid phase and fluid phase concentrations of the highly radioactive radionuclide Tc-99 are many orders of magnitude higher under oxidizing conditions compared to reducing conditions, it would be expected that releases from the saltstone would be sensitive to the rate of formation of the oxidized layer.

The other primary potential release mechanism occurs because of flowing water within a fracture. A certain amount of water will imbibe from the fracture into the saltstone matrix when the matrix has a capillary suction less than necessary to maintain flowing water in the fracture; hence, releases will be due to radionuclides diffusing against the direction of water movement. However, the fracture also provides a potential pathway for oxygen to move into the saltstone matrix so that oxidizing conditions may be found at the fracture surface.

In the PORFLOW simulations of release from the saltstone and vaults, DOE used moisture characteristic curve data in the description of moisture flow under unsaturated conditions (WSRC, 2005b). In the RAI, NRC staff questioned the use of the data because of anomalously high increases in gas flow during the laboratory test (NRC, 2005a). Upon re-examination of the data, DOE acknowledged that the testing cannot be used as the basis for moisture characteristic curves for the numerical modeling. DOE assessed the impact of the use of the questionable data by assuming the relative permeability of the saltstone and concrete was 1.0 regardless of the saturation, and found that dose increased by a factor of 4 over the base case results (WSRC, 2005a). The NRC believes the sensitivity analysis performed to address this concern was bounding, but that either the impacts should be considered as part of the base case or additional tests should be performed to replace the questionable data.

DOE’s demonstration that the disposal system could meet the performance objectives was largely determined by several factors, including: (1) water passing through the saltstone matrix is limited; (2) radionuclide diffusion rates are limited; (3) water passing through fractures is limited; and (4) oxidation near saltstone boundaries is limited.

The dose significance of water flowing through the saltstone matrix is clearly demonstrated in several sensitivity scenarios modeled by DOE (sensitivity scenarios 12, 23, 24, 25, 27, and 28 in WSRC, 2005a), in which increased mean annual precipitation and increased vault saturated hydraulic conductivities have a large influence on dose even when oxidation does not occur. Water flowing through the matrix determines advective releases from the vault together with radionuclide concentrations at the edge of the vault where the water exits. Based on scenarios 4 through 11 (see Table 8), dose uncertainty is more sensitive to uncertainty in the saltstone hydraulic properties than in the concrete properties, as might be expected from the much larger path length within saltstone. However, the vault and saltstone wasteform act independently as barriers to moisture movement in the DOE modeling, such that the largest increases in dose are observed for scenarios when both barriers are degraded. Increasing the mean annual precipitation by 25%, which is not unreasonable for representing unexpectedly poor closure-cap performance, increases doses by a factor of 23. The NRC staff interpretation of the sensitivity results is that the water fluxes removing radionuclides by advection are largely determined by (1) hydraulic properties of the inner core of the saltstone and to a lesser extent the concrete vault and (2) closure-cap performance. In other words, if the inner core remains substantially undegraded, advective releases would be relatively insensitive to degradation of the concrete and outer fringe of the saltstone, since most of the lower exit boundary would be under a low-permeability umbrella. Degradation of the inner core is difficult to predict because of a lack of data about the long term performance of cementitious materials. However, the NRC staff believes that degradation is likely to proceed to a greater extent near the accessible environment.

The dose significance of radionuclide diffusion is examined in scenarios 14 through 18 (WSRC, 2005a), in which an order-of-magnitude increase in the diffusion coefficient in both concrete and saltstone increases doses by a factor of 14, while increasing either coefficient independently has a much smaller response. The diffusion coefficient value used for soil, while not expected to influence transport significantly, is at the low end of values reported for common cations and anions in water from the cited reference (Domenico and Schwartz, 1990), reasonably accounting for tortuosity effects. It appears that a unit conversion error may have reduced the soil value by an order of magnitude in DOE's simulations. Table 4-6 in (Rosenberger et al., 2005) provides a molecular diffusion coefficient used in the modeling of 158 cm^2/yr when the correct value based on 5×10^{-5} cm^2/s should be 1580 cm^2/yr. It is not believed that the molecular diffusion coefficient of soil is a key factor in estimating release rates to the saturated zone. The molecular diffusion coefficients used for concrete and saltstone were 4 orders of magnitude smaller than the range reported in the cited reference, but DOE leaching experiments and external literature suggest that the diffusion coefficients are reasonable for intact concrete. Additional diffusive behavior occurs due to mechanical dispersion of moving fluids, which would also potentially increase diffusive releases. DOE does not justify the neglect of mechanical dispersion, but the NRC staff notes that mechanical dispersion is more significant as length scales and water velocities increase, and therefore should not be significant over release-significant scales on the order of a meter with the slow velocities expected within the saltstone and concrete. In scenarios where the saltstone and vault were assumed to deteriorate hydrologically, DOE did not consider an increase in the diffusion coefficient over time as the concrete vault and saltstone degrade or develop cracks. Cracking would be expected to significantly shorten the diffusive path length and could result in increases in the effective diffusion coefficient depending on the frequency and severity of the cracking. Since both the effective diffusion coefficient and hydraulic conductivity are intimately related to pore structure in cementitious materials, it is reasonable to expect that both coefficients would be subjected to some increase over time.

The significance of water flowing through fractures is examined in scenarios 31 and 32, in which three discrete features are included in the simulation as vertical strips of saturated gravel, with both fractures and matrix assumed to have artificially saturated hydraulic conductivity in order to maximize flows. When completely reducing conditions were assumed, doses increased by a factor of 73 (essentially at the regulatory limit when results are appropriately scaled. See Section 4.2.16), while assuming completely oxidized conditions increased doses a further factor of 7 because of an overall factor of 810 increase in Tc-99 releases. An additional scenario completed by DOE but not provided in their table of sensitivity scenarios simulates the effect of filling the fracture with 2.5 cm (1.0 in) of native soil, showed a factor of 23 increase in dose (without oxidation). These results should be comparable to filling the fracture with the gravel drainage layer material, since the two materials would have similar hydraulic conductivity above 1,000 cm (33 ft) of suction head. DOE dismisses fracture flows as not credible because of an analysis that indicates fracture flows would be negligible if the lower drainage layer does not develop sufficient saturation such that there is a head difference to overcome the air entry pressure of the fractures. The NRC staff agrees with the approach of the analysis, but disagrees with the interpretation because of the limited consideration of a number of sources of uncertainty and limited model support for the numerical predictions.

Model support is needed to justify the model results that flow in fractures in the vaults and saltstone will not occur in the unsaturated system. A number of real world observations suggest that moisture will flow in joints or fractures under conditions that may not be predicted by numerical models. For example, observations at the tank farm facility at a DOE facility in Idaho suggest that water flows from unsaturated soils through joints in the vaults surrounding the high-level waste tanks (Lockie, 2002). It is postulated that flow occurred primarily as a result of a rapid influx of snow melt or precipitation events. While the system at SRS may not be completely analogous, there are many similarities and the observations suggest that the temporal and spatial averaging used in the calculations must be consistent with the expected real-world system. Modeling to justify the lack of flow through fractures would typically include the following features: (1) heterogeneity in material properties; (2) temporal variations in saturation, especially resulting from rapid transport of infiltration through root holes to the lower drainage layer; (3) infilling of fractures and joints with porous media; (4) offset of the essentially impermeable vault/saltstone on either sides of the fractures; (5) variability in the aperture of the joints and fractures; (6) sensitivity to grid size in the simulations, especially at the interface of the fracture and porous media; and (7) variability in moisture characteristic curve parameters. Ideally, this type of analysis would be performed probabilistically because of the many sources of uncertainty. The modeling results must be supported by appropriate information, such as a blind prediction of observations at analogous systems, lab- and field-scale experiments, or other forms of model support that take into account site-specific behavior.

The dose significance of the oxidation state near matrix boundaries is examined in scenarios 20–22, 25, 29, 30, and 33 (WSRC 2005a). The only radionuclide examined by DOE that is significantly affected by the oxidation state is Tc-99, with negligible doses under reducing conditions. Under the assumption of complete oxidation but otherwise nominal conditions, there was a factor of 67 increase in dose relative to the reduced state and the calculated dose was very close to the regulatory limit. Under wetter conditions (scenarios 25 and 30), the assumption of complete oxidation resulted in a factor of 180 increase to dose relative to the reduced state, with failure to meet the regulatory limit (when results are appropriately scaled. See Section 4.2.16) regardless of the oxidation state. Under highly degraded conditions (scenario 33), complete oxidation results in a factor of 110 increase to dose relative to the reduced state. The NRC staff agrees with the following conclusions made by DOE: (1) it is unlikely that more than a fraction of the saltstone mass will oxidize during the performance period, (2) the oxidized zone will likely be where the saltstone is near the accessible environment and along fractures, and (3) calculations with more representative oxidation conditions will likely fall between the two extremes. The DOE interpretation is that the bulk saltstone matrix, essentially unaffected by oxidation, will govern Tc-99 releases, while the NRC staff interpretation is that a thin rind of oxidized saltstone may greatly increase Tc-99 releases relative to the DOE conceptual model.

The sensitivity simulations suggest that Vault 4 performance is largely controlled by: (1) closure-cap performance eliminating fracture flow and reducing matrix flow; (2) degradation rates in the inner core of the saltstone matrix increasing hydraulic conductivity; (3) degradation rates in the outer fringe of the saltstone matrix and concrete vault increasing diffusion and advection; and (4) oxidation of the saltstone near boundaries with the accessible environment. Degradation and oxidation of the concrete and saltstone are discussed further in Section 4.2.7.

The strategy followed by DOE is to use a series of steady-state flow fields to calculate releases and subsequent transport, apparently without adjusting the fluid densities to account for the dissolved salts within the fluid. Using steady-state flow fields is readily justified if the entire flow system equilibrates quickly to changes in boundary conditions relative to the period of analysis. The saltstone matrix has not been demonstrated to respond quickly, and it would be expected that response times could easily be measured in thousands of years with the nominal properties used by DOE. If the suction head in the saltstone matrix is initially much greater than the steady state condition (the matrix is drier than calculated), it may be conservative to use a series of equivalent steady-state flows because capillary effects will tend to draw water into the saltstone, thereby reducing releases relative to the steady-state calculation. Alternatively, releases could be enhanced if the matrix is initially wetter than the steady-state flow estimates since water in the matrix must leave to achieve the long-term conditions. There are no measurements of the initial conditions in the saltstone. The neglect of fluid density (e.g., brine) in the flow calculations may also affect release rates and transport in the vadose zone. PA calculations could account for brine releases simply by performing transient simulations with coupled density-dependent flow and salt transport, where the modeled salt consists of the suite of dissolved constituents. This could automatically account for the slow response time of the saltstone and concrete matrix.

4.2.10 Hydrology and Far-Field Transport

For the purposes of this review, the far field is defined as that portion of the contaminant transport pathway that starts at the water table. Near-field and vadose zone processes are discussed in previous sections of this report, but discussion of K_d's used for vadose zone transport will be included in this section.

In the recent special analysis, DOE modeled contaminant groundwater transport from the water table below Vault 4 to the compliance boundary 100 m down-gradient using the flow and transport code PORFLOW (Cook, et al., 2005). The localized, steady-state groundwater model used for Vault 4 was based on the regional model described by Flach (2004). Site descriptive information, including hydrogeology, is detailed by Cook and Fowler (1992, Section 2). The aquifers affected by contaminants from Vault 4 in the model are largely unconsolidated, Tertiary Coastal Plain sands with variable clay contents and some clay layers. In the immediate vicinity of Vault 4, the water table is approximately 23 m (75 ft) below the planned ground surface above the vault and approximately 12 m (40 ft) below the bottom of the vault. For PORFLOW, a "reduced" local model, based on the larger regional model of Flach (2004), was constructed for a 640- by 580-m (2,100- by 1,900-ft) area around Vault 4. A horizontal grid with 30 m (100-ft) elements was constructed and vertical grid elements of variable height were selected to reflect hydrostratigraphy (Cook, et al., 2005, Section A.3). Compliance nodes that intersect the groundwater zones most affected by Vault 4-derived contaminants were assigned for monitoring model contaminant concentrations. For radionuclides, these nodes were around 100 m (328 ft) to the east-northeast (i.e., down-gradient) of the Vault 4 footprint. Flow modeling showed that contaminants from Vault 4 will be restricted to the lower unit of the Upper Three Rivers Aquifer in the flow path to the 100-m (328-ft) compliance boundary; therefore, compliance nodes were restricted to that unit. The underlying Gordon Confining Unit will prevent transport to the Gordon Aquifer.

The inputs to the saturated zone PORFLOW model consist of time histories of contaminant fractional release rates from the vadose zone model. The release rate, which is normalized to the source concentration, is recalculated as a concentration term based on the water volume of the model source nodes. For radionuclides, the model output is fractional concentration in units of pCi/L/Ci of source inventory in groundwater at each of the compliance nodes. For demonstrating compliance with the 10 CFR 61.41 performance objectives, DOE uses the PORFLOW outputs to calculate groundwater doses from multiple pathways (Rosenberger, et al., 2005). Model doses (per curie of source radionuclide) are based on peak groundwater concentrations, conservatively combining all peak radionuclide concentrations for the dose calculation regardless of when they occur.

The key parameters affecting model radionuclide transport are saturated hydraulic conductivity (hydrostratigraphic unit specific) and sorption coefficient, or K_d (radionuclide-specific). Saturated zone hydraulic conductivity values for the special analysis were based on characterization data and calibrated to well data (Flach and Harris, 1999; Flach, 2004, Figure 2-6). Model sorption coefficients with references were reported in Rosenberger, et al. (2005, Table 4-12) and further discussed was provided in Westinghouse Savannah River Company (2005b, pp. 277–292). For vadose zone (below the disposal vault) and saturated zone transport modeling, tabulated soil values were used for all units except the Gordon Confining Unit, for which clay values were used (Pickett, 2005). For many radionuclides, the literature compilation for sand soils by Sheppard and Thibault (1990) was cited. For all other radionuclides (except tritium and californium, which are risk-insignificant), site-specific information was used in developing sorption coefficients. For plutonium DOE adopted an oxidation state model that distributed plutonium isotopes into two species representing less mobile Pu (III+IV) and more mobile Pu (V+VI). The ratio between the species was based on field measurements of oxidation and reduction rates.

The PORFLOW model calculated short groundwater travel times from the water table below Vault 4 to the compliance nodes of less than 10 years (Cook, et al., 2005, Figures A–40 and A–42). For the special analysis base case, fractional concentration time histories at the 100-m (328-ft) compliance boundary for radionuclides are presented in Figures A–47 through A–76 of Cook, et al. (2005). All concentration values are very low and, except for radionuclides completely depleted by decay, rise at the end of the 10,000-year simulation period. The latter observation is true even for radionuclides such as Tc-99 that have very low soil K_d's and should therefore travel very quickly through the saturated zone. This relationship demonstrates that the saturated zone does not provide an effective barrier for relatively poorly sorbing radionuclides.

DOE (Cook, et al., 2005; Rosenberger, et al., 2005; WSRC, 2005a,b) provided limited sensitivity studies directed at saturated zone parameters. Recharge and hydraulic conductivities of the Gordon Confining Unit were varied in WSRC (2005b, pp. 270–276) to represent uncertainty in saturated zone flow rates; modest sensitivity to dose was found (a dose increase by a factor of 1.7 for the worst case). No analyses were sensitive to K_d in isolated subvault vadose or saturated zones. Sorption coefficients in vadose and saturated zone natural media were decreased in two of the recently reported sensitivity studies (WSRC, 2005a, scenarios 19 and 26), but the values also were decreased in saltstone and concrete in the same analyses. However, there is indirect evidence that model results are much more sensitive to saltstone and vault characteristics than to geosphere sorption coefficients (WSRC, 2005a). For example in scenario 26, reduction of K_d values in all media throughout the model system by a factor of ten increased dose by a factor of approximately 40. However, cases in which the Tc-99 K_d in the saltstone and vault alone was decreased due to oxidizing conditions (scenarios 21, 22, and 29) yielded far greater dose increases, as did cases involving saltstone hydraulic properties and oxidizing conditions (scenarios 30, 32, 33). Because (1) Tc-99 is responsible for these calculated dose increases, (2) the natural system Tc-99 K_d is already low (0.1 mL/g), and (3) saturated zone hydraulic and sorption characteristics are better constrained (i.e., uncertainty is reduced) than the equivalent long-term parameters for the disposal system, dose is not apparently highly sensitive to a reasonable range of saturated zone transport characteristics in this model system.

Another indication of saturated zone performance can be seen in the sensitivity analysis for water usage inside the 100-m (328-ft) boundary (Rosenbereger, et al., 2005, Section 8.3.3). In this analysis, peak vadose zone annual radionuclide releases to the water table were diluted into the top mesh layer of the saturated zone and not subjected to transport and retardation. This water volume formed the basis for an all-pathways groundwater dose calculation; the dose was mainly due to Se-79. For comparison to the base case groundwater pathways analysis (Rosenberger et al., 2005, Section 4.1), which used a different inventory, it is convenient to look at dose per activity values for Se-79. The sensitivity case neglecting saturated zone transport yielded a value of 3.5×10^{-14} mSv/Bq-yr (0.13 mrem/Ci-yr) for Se-79, whereas the base case value was 6.8×10^{-14} mSv/Bq-yr (0.025 mrem/Ci-yr). In other words, unrealistically neglecting any retardation and attenuation effects of the saturated zone increased dose by only a factor of five. In light of sensitivity studies of vault and saltstone parameters (WSRC, 2005a), this effect is minimal. Neglecting retardation and attenuation in the saturated zone had a small effect because those radionuclides with low natural system K_d values (e.g., Tc-99, Se-79, I-129)-and which are therefore less affected by saturated zone transport-also tend to dominate dose in the base case (or in disposal system sensitivity cases for Tc-99) because of their low K_d values in the disposal system.

4.2.11 NRC Evaluation – Hydrology and Far-Field Transport

The PORFLOW implementation of groundwater flow and transport modeling for Vault 4 is well supported, appropriately integrated with regional and local hydrogeological conditions, and well suited for the purpose. NRC staff note, however, that the model may be overpredicting the ability of the saturated zone to dilute the vadose zone-derived contaminant plume. According to WSRC (2005a, Figure 18-2), the concentration of relatively unretarded I-129 decreases by a factor of greater than 1×10^5 during transport from the vadose zone to the 100-m (328-ft) compliance location. Figures A–43 and A–44 of Cook, et al. (2005) show that horizontal fluxes in the saturated zone are much greater than the vertical influxes from the vault, which staff agrees will result in dilution of the concentrations. However, the model grid layers in the lower aquifer above the Gordon Confining Unit are about 6-m (20-ft) thick vertically, while particle traces from adjacent corners show a vertical extent of about 1.2 m (4 ft) (Cook, et al., 2005, Figure A–43). The actual plume would be expected to be similar to the envelope of particle traces, while in numerical simulations radionuclide mass is smeared over one or more elements in the vertical dimension. Because PORFLOW calculates average concentration within a grid cell, a simple mass balance suggests that peak concentrations are diluted by at least a factor of five due to the coarse vertical grid resolution. A smaller vertical grid dimension would result in significantly higher concentrations being calculated in certain cells, rather than diluting the contaminants throughout a larger volume. Similar considerations apply in the horizontal dimension. A finer model grid may give more realistic contaminant concentrations. A reduction in the grid spacing in the portion of the model immediately surrounding Vault 4 and the compliance nodes to verify that peak concentrations are properly calculated should be considered. In practice, grid refinement might be accomplished by using a separate nested grid aligned with the groundwater flow direction for transport calculations while using boundary conditions from the regional grid. Alternately, DOE can ensure that assumptions regarding dilution from withdrawals in a well would result in dilution effects as large or larger than that introduced from the grid refinement.

As described on page A-54 of Cook, et al. (2005), the model vadose zone contaminant flux is distributed evenly among the 12 source nodes at the water table. New sensitivity analyses have considered the possibility of release and transport along large-scale fractures in the saltstone and vault (WSRC, 2005a, scenarios 31 and 32). Focused release and transport of contaminants from the disposal system could result in more localized pulses than would be represented by the vault footprint. Averaging these pulses over the footprint is another potential source of dilution. If further analyses were to conclude that large-scale fracture flow through the disposal system were credible, the horizontal resolution of the model grid should be reevaluated.

DOE performed a sensitivity analysis on plume interaction between Vaults 1 and 4 that suggested only a 25% increase in dose from inclusion of the Vault 1 plume (Cook, et al., 2005, Section 7.5.1). That report also mentioned further consideration of plume interactions in future PAs. These considerations should include all planned disposal systems and should consider effects beyond the compliance location for any one vault. For example, the dose from two intersecting plumes farther than 100 m (328 ft) from the two sources could potentially exceed the dose at any single 100-m (328-ft) compliance location.

Saturated hydraulic conductivities used in the model are based on extensive characterization and well calibration. Calculated dose is not strongly sensitive to these parameters. The PORFLOW model appears to have appropriately bounded any uncertainty in hydraulic conductivities in the saturated zone.

As discussed in the previous section, calculated dose is less sensitive to order-of-magnitude changes in K_d than to changes in Tc-99 K_d alone due to oxidizing conditions. In addition, the Tc-99 K_d for the vadose and saturated natural systems is so low that lower values will not strongly affect dose. Therefore, a risk-informed evaluation of sorption coefficients will address uncertainties or variability on an order-of-magnitude scale. It appears that, in most cases, DOE has appropriately chosen K_d values for natural system transport simulations. Where site-specific data were lacking, literature values that tended toward the low end of ranges were selected. For example, in Rosenberger, et al. (2005), model soil values were chosen from the sand data in Sheppard and Thibault (1990), instead of from data more representative of more sorptive loam, clay, and organic soils. In addition, in nearly every case, a selected site-specific value was lower than the corresponding literature value. A notable exception is uranium, for which a K_d of 800 mL/g was selected from pH-dependent, field-derived concentration ratios between soils and coexisting waters (Serkiz and Johnson, 1994). This value is much higher than the Sheppard and Thibault (1990) sand value of 35 mL/g and the value of 40 mL/g used in some previous SRS models (Serkiz and Johnson, 1994). A number of uncertainties are attendant on application of concentration ratios as such K_d values, including the uranium sorption mechanism (e.g., precipitation versus surface complexation); redistribution during sampling, storage, and handling; and applicability of F and H seepage basin conditions to groundwater flow in the Z-area subsurface. These uncertainties, in light of the lack of more site-specific data obtained under better-controlled conditions, suggest that 800 mL/g may not be suitably conservative. Under current model conditions, the K_d for uranium does not appear to be risk-significant; however, if the PA is significantly changed, the uranium K_d should be reassessed.

Aquifer geochemical conditions can directly affect radionuclide sorption. In the current model, release of contaminants to the saturated zone appears unlikely to significantly change saturated zone water chemistry. This is because of the apparently high degree of dilution afforded by the saturated zone (WSRC, 2005a), such that, nitrate concentrations and pH would not change appreciably due to the chemical plume emanating from the disposal system. However, DOE has not provided an analysis of this potential effect. If subsequent calculations suggest that dilution at the water table may be less than previously predicted, potential chemical effects on saturated zone sorption coefficients should be considered. A straightforward means for assessing this issue would be a sensitivity analysis that lowers only chemically sensitive saturated zone K_d values to appropriate levels. In addition, available information does not address the potential for vadose zone sorption to be less effective than expected due to contaminant plume chemistry. Again, this question could be addressed by a sensitivity analysis focused only on vadose zone natural material K_d values.

4.2.12 Dose Methodology

The dose methodology used by DOE in the SDF PA process was the application of dose conversion factors. This methodology is widely used in PAs and consists of multiplying the radionuclide concentration in air, water, or soil that a receptor might be exposed to through any of the various pathways by the dose conversion factor specific to that ingestion or inhalation process and radionuclide.

The groundwater pathway dose calculation was conducted by DOE using the LADTAP XL© spreadsheet code (Simpkins, 2004) discussed previously. The dose factors used in LADTAP XL© are those specified by DOE in PAs conducted throughout the DOE complex in two separate documents on external and internal dose conversion factors for calculation of dose to the public (DOE, 1988a,b). The calculation process and the dose factors for the groundwater pathway PA used are described in Cook, et al. (2005); Rosenberger, et al. (2005); and Jannik (2005).

The air pathway dose calculation is conducted within the widely used EPA computer code CAP88 (EPA, 2002). CAP88 was used to calculate dose factors in units of mrem/yr per Ci/yr released at the ground surface, Savannah River Site boundary, and 100 m (328 ft) from Vault 4 (Cook, et al., 2005). This dose conversion factor methodology has been used in other PAs conducted by DOE, EPA, and NRC.

In the intruder analysis performed by DOE for the SDF PA, radionuclide dose conversion factors from the Federal Guidance Reports developed by EPA (EPA, 1993, 1988) were used (Cook, et al., 2005; Rosenberger, et al., 2005). Ingestion and inhalation dose conversion factors were taken from Federal Guidance Report No. 11 (EPA, 1988), and external dose conversion factors were taken from Federal Guidance Report No. 12 (EPA, 1993). Both of these reports provide 50-year committed effective dose equivalents per unit of activity based on the exposure pathway (inhalation, ingestion, or external) and the specific radionuclide. The intruder analysis dose conversion factors were included in the intruder analysis code developed by DOE (Koffman, 2004).

4.2.13 NRC Evaluation – Dose Methodology

The dose methodology implementation of the SDF PA is well supported and suited for the purpose. Numerous NRC guidance documents provide recommendations on the approach and use of the specific dose conversion factors used in the SDF PA process. These include NUREG–1573 (NRC, 2000b), which provides guidance on the use of pathway dose conversion factors for calculating doses via the potential exposure pathways, and NUREG–1757 (NRC, 2003b, Volume 2, Appendix I), which provides guidance on the use of specific dose conversion factors such as those developed by the EPA and published in Federal Guidance Report Nos. 11 and 12 (EPA, 1988, 1993).

4.2.14 Overview of Performance Objectives

The NDAA establishes the applicable criteria for determining that waste is not HLW, including that the waste will be disposed of in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations. The performance objectives provide criteria to ensure that the public, workers, and environment will be protected from releases of radioactivity.

10 CFR 61.41, "Protection of the general population from releases of radioactivity," 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." (NRC, 2001a)

The 0.25-mSv/yr (25-mrem/yr) limit applies for the post-closure period of a disposal facility. The other radiological control limits of 10 CFR Part 20, "Standards for Protection Against Radiation," apply during facility operation (NRC, 2001b).

10 CFR 61.42, "Protection of individuals from inadvertent intrusion," 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."

Although a particular dose limit is not specified in this performance objective, compliance with the technical requirements of Part 61 and, in particular, with the classification system of 10 CFR 61.55, is considered to provide adequate protection to intruders at a near-surface land disposal facility. In the Draft Environmental Impact Statement for Part 61 (NRC, 1981), NRC used a 5-mSv (500-mrem) dose limit to an acute inadvertent intruder to establish the concentration limits and other aspects of the waste classification system. In addition, Part 61 does not specify a time for institutional controls in the performance objectives, but does require, in 10 CFR 61.59(b), that "... controls may not be relied upon for more than 100 years."

10 CFR 61.43, "Protection of individuals during operations," 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 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable."

This performance objective applies to both the public and to disposal facility workers.

10 CFR 61.44, "Stability of the disposal site after closure," 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 stability performance objective is consistent with a premise of Part 61 that the facility must be sited, designed, used, operated, and closed with the intention of providing permanent disposal. A disposal facility should not require long-term maintenance and care. Stability is particularly important considering the requirements in 10 CFR 61.59(b) that "... institutional controls must not be relied upon for more than 100 years following transfer of control of the disposal site to the owner."

Table 7 provides an overview of the DOE results compared to the performance objectives of 10 CFR 61, Subpart C. The results are explained in more detail in the sections that follow.

4.2.15 Protection of the Public

The public is represented by an adult member of a farming community (e.g., resident farmer) that lives in a residence downgradient of the SDF. During the operational and institutional control periods, it is assumed that the individual resides at the SRS site boundary. After active institutional controls cease at 100 years, the member of the public resides 100 m downgradient from Vault 4. As previously indicated, up to 14 additional vaults would be needed to dispose of saltstone grout. Groundwater modeling to assess the impact from all of the vaults considering the potential for overlap of plumes from individual vaults at any potential receptor location was

Table 7 Summary of DOE Results Compared to Performance Objectives

Part 61 Performance Objective	Performance Limit	DOE PA Result (mrem/yr)
61.41 All-pathways dose to public	25 mrem/yr	2.3[2]
61.42 Intruder - Resident scenario	500 mrem/yr	21
61.42 Intruder - Agricultural scenario[1]	500 mrem/yr	149
61.42 Intruder - Post drilling scenario[1]	500 mrem/yr	31
61.43 Protection of individuals during operations	5 rem/yr[3]	NA[4]

(1) Because of the long-term performance of the engineered cap to prevent erosion and the long-term durability of the saltstone wasteform, DOE believes these scenarios are not credible intruder scenarios. Results were provided for these scenarios as part of sensitivity analysis.
(2) This result is for all of the current and projected radiological inventory being placed in one vault.
(3) There are numerous numerical limits for protection of individuals during operations, the value shown is the limit for protection of workers.
(4) NA = not applicable. Not calculated in the DOE PA.

not completed at the time the draft waste determination was submitted. Therefore, to demonstrate conformance with the performance objectives, DOE assumed that all of the radiological inventory expected to be distributed over the future vaults was located within the existing Vault 4. A member of the public is assumed to use water from a well for domestic purposes following the institutional control period. The well is assumed to be located where the maximum concentration of radionuclides in the groundwater is predicted to occur. Modeled contaminant concentrations are not diluted as a result of clean water entering the well if the extraction of contaminated water from the well exceeds the amount of contaminated water available. The groundwater pathway calculations were performed in two steps. First the flux to the water table for parent radionuclides and progeny were estimated with vadose zone flow and transport simulations. Changes to properties of the engineered cap were estimated through additional analysis and input to the vadose zone flow and transport calculations. Using the estimated fluxes from the vadose zone modeling, saturated zone flow and transport modeling was used to estimate the groundwater concentrations at a hypothetical well. It is assumed that contaminated water from the well is used for the following purposes: (1) drinking water, (2) pond water (in which fish are raised and recreational activities occur), and (3) irrigation for growing vegetables and raising livestock to provide meat and milk. The gaseous flux of radionuclides to the land surface over time was estimated by DOE and used to calculate the doses from direct plume shine, inhalation, and ingestion of vegetables, meat, and milk exposed to airborne radioactivity. For most radionuclides the dominant pathways were water consumption, vegetable consumption, and meat and milk consumption; however, important pathways may be radionuclide- and scenario-specific. The dose assessment was more complex than summarized here (Cook and Fowler, 1992).

The dose from all radionuclides that met the screening criteria and all pathways was computed by assuming that the maximum concentration in contaminated groundwater and air occurred at the same time for each radionuclide. The whole body dose to a member of the public was estimated to be 0.023 mSv/yr (2.3 mrem/yr) over the 10,000 year analysis period, which does not exceed the 10 CFR Part 61 limit of 0.25 mSv/yr (25 mrem/yr) to the whole body.[1] Over 99 percent of the dose was from Se-79 and I-129, principally from the ingestion pathway. EPA (1988) values for ingestion dose conversion factors were used to determine the 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 (Rosenberger et al., 2005). The results indicated that the dose to the thyroid would be 0.046 mSv/yr (4.6 mrem/yr) and the dose to any other organ would be 0.053 Sv/yr (5.3 mrem/yr), which are below the performance objectives of 0.75 mSv/yr (75 mrem/yr) and 0.25 mSv/yr (25 mrem/yr) for the thyroid and any other organ, respectively.

As indicated in Section 4.2.1, DOE's PA analysis was deterministic. Initially, limited sensitivity analysis was performed to evaluate the impact of uncertainty and to identify key parameters or conceptual models. In response to the RAI and subsequent meetings with NRC staff, DOE completed an expanded sensitivity analysis to address a number of parameters, including combinations of parameters that may represent alternative scenarios differing from DOE's base case scenario (WSRC, 2005a). NRC staff had suggested that the sensitivity cases be defined such that they would provide an understanding of how reliant the dose results were on various parameters, models, or assumptions, with special emphasis on the engineered cap and vault/wasteform. Thirty five sensitivity cases were run by DOE and the main areas evaluated were: (1) infiltration through the engineered cap, (2) the hydraulic conductivities of the vault and saltstone and the rate at which they degrade, (3) the effective diffusion coefficients of the saltstone and vault, (4) the impact of fracture flow through cracks in the vault, (5) the saturation state of the vault and saltstone, and (6) the oxidation state of the vault and saltstone.

The SDF is a controlled release facility, such that proper closure to meet the objective of limiting moisture through the waste and to limit deterioration of the wasteform is an integral part of the long-term acceptability of the salt waste management strategy. DOE's original base case was based on the assumption that essentially no deterioration of the engineered cap or wasteform over the 10,000 year analysis would occur; therefore, water available to the waste was limited to 0.175 cm/yr (0.0689 in/yr). This small amount of infiltration represented less than half a percent of projected precipitation, whereas for many vegetated humid sites natural recharge values are commonly in the range of 15 - 40% of precipitation. Because the engineered components of the system did not degrade, releases were limited to diffusive releases and the doses to the public receptor through the groundwater pathway were limited to 1×10^{-5} mSv/yr (0.001 mrem/yr)(Cook and Fowler, 1992). It should be noted, that the result from 1992 (i.e., 1×10^{-5} mSv/yr [0.001 mrem/yr]) is much lower than the current result (0.023 mSv/yr [2.3 mrem/yr]) because of three primary factors: (1) more degradation is assumed in the current analysis, (2) the inventory in the current analysis is much larger, and (3) the current analysis is an all-pathways dose assessment compared to the previous analysis which only calculated the dose from drinking water ingestion. DOE's revised base case did consider deterioration of the engineered system (e.g., engineered cap and vault/wasteform) over time (WSRC, 2005a).

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[1] The dose methodology used in 10 CFR Part 61, Subpart C [based on International Commission on Radiological Protection Publication 2 (ICRP 2)], is different than that used in the newer ICRP 26. However, the resulting allowable doses are comparable, and DOE-SRS used the newer methodology in ICRP 26.

The current closure concept has a geosynthetic cover system. After the institutional control period, infiltration is predicted to gradually increase over time as the closure system degrades, primarily from the intrusion of deep-rooted plants (e.g., pine trees) and siltation of drainage layers (e.g., from colloidal clay migration). The engineered cap has an upper GCL and a lower GCL. The upper GCL was assumed to degrade over time by pine tree penetration (Phifer and Nelson, 2003). Root penetration of the GCL by pine trees was simulated to result in holes through the GCL that would act as direct water pathways for infiltration from the upper drainage layer to the lower backfill layer. After precipitation events, saturated conditions were predicted to occur around the holes such that a radius of influence would form that would be much greater than the radius of the hole itself. As a result, a small area of holes in the GCL can greatly reduce the lateral flow of water in the drainage layer and increase the vertical flow of water into the lower backfill. Holes covering only approximately 0.3% of the GCL resulted in infiltration near that of typical background infiltration. DOE estimated that infiltration through the upper GCL would be 1.6 cm/yr (0.63 in/yr) shortly after closure, increase to approximately 35 cm/yr (14 in/yr) by year 2000 after closure, and would remain relatively constant until year 10,000. Alternate land use scenarios resulted in a range of infiltration rates after degradation of the upper GCL (i.e., 16 to 54 cm/yr [6.3 to 21 in/yr]).

As discussed in Sections 4.2.6 and 4.2.8, the vault and saltstone are designed to have a very low hydraulic conductivity to greatly limit releases into the natural system and eventually into the groundwater aquifer. Based on tests performed at Core Laboratories, the hydraulic conductivity of the vault and saltstone are assigned values of 1×10^{-12} cm/s (1×10^{-6} ft/yr) and 1×10^{-11} cm/s (1×10^{-5} ft/yr), respectively (Yu et al., 1993). Fly ash is added to the saltstone to lower the redox potential of the pore water, thereby effectively eliminating the risk from Tc-99 if reducing conditions are maintained. In common cementitious systems, Tc-99 has a K_d of 1000 ml/g under reducing conditions but a K_d of approximately 1 ml/g under oxidizing conditions (Bradbury and Sarott, 1995). Instead of attempting to predict the long-term evolution of the hydrologic properties of the vault and saltstone, DOE used professional judgment to assign degraded values for hydraulic conductivity of the vault and saltstone of 1×10^{-9} cm/s (1×10^{-3} ft/yr) at 10,000 years. The timing for the increase in hydraulic conductivity was implemented as multiple step functions in the PA. Sensitivity analysis was used to identify the importance of the assigned value of the degraded hydraulic conductivity to meeting the performance objective.

Table 8 provides a summary of sensitivity runs performed by DOE, and Table 9 is a companion table that provides a more detailed description of each analysis. The results in Table 8 are for the current and projected Vault 4 inventory based on DOE's proposed approach to waste treatment (WSRC, 2005b). As previously indicated, 35 sensitivity scenarios were run. The main areas evaluated were: infiltration through the engineered cap, the hydraulic conductivities of the vault and saltstone and the rate at which they degrade, the effective diffusion coefficients of the saltstone and vault, the impact of fracture flow through cracks in the vault, the saturation state of the vault and saltstone, and the oxidation state of the vault and saltstone. Because the analysis approach was deterministic, combinations of uncertainties were evaluated in order to better assess their impact. For the first nineteen sensitivity runs performed, four radionuclides were evaluated: H-3, C-14, Se-79, and I-129. The additional sixteen runs had additional radionuclides evaluated depending on the goal of the analysis. The additional radionuclides considered were: Tc-99, U-238, Np-237, Pu-238, Pu-239, Cs-137 and Sr-90, although not all of these radionuclides were considered in each simulation. The radionuclides considered by DOE were limited because of the additional effort required to verify the accuracy of the results.

Table 8 Summary of DOE Sensitivity Analyses for Vault 4

Scenario	Description	Result (mrem/yr)	Radionuclides-Largest Contributors
1	Base case	0.048	Se-79, I-129
2	Optimistic cover degradation	0.02	Se-79
3	Pessimistic cover degradation	0.27	Se-79, I-129
4	Optimistic vault degradation	0.031	Se-79
5	Pessimistic vault degradation	0.051	Se-79, I-129
6	Optimistic initial vault conductivity	0.032	Se-79
7	Pessimistic initial vault conductivity	0.051	Se-79, I-129
8	Optimistic saltstone grout degradation	0.037	Se-79
9	Pessimistic saltstone grout degradation	0.23	Se-79, I-129
10	Optimistic initial saltstone grout conductivity	0.038	Se-79
11	Pessimistic initial saltstone grout conductivity	0.24	Se-79, I-129
12	Pessimistic cover, vault, grout degradation	4.0	I-129, Se-79
13	Vault and grout saturated	0.18	Se-79, I-129
14	Optimistic vault diffusion	0.035	Se-79, I-129
15	Pessimistic vault diffusion	0.17	Se-79, I-129
16	Optimistic saltstone grout diffusion	0.039	Se-79, I-129
17	Pessimistic saltstone grout diffusion	0.064	Se-79
18	Pessimistic vault/saltstone grout diffusion	0.67	Se-79, I-129
19	Pessimistic K_d values	36	C-14, Se-79, I-129
20	Reducing conditions in vault at all times. Only Tc-99 was considered in the analysis	1.6×10^{-13}	Tc-99
21	Oxidizing conditions (K_d Tc-99 = 1 mL/g). Only Tc-99 was considered in the analysis	3.2	Tc-99
22	Oxidizing conditions (K_d Tc-99 = 0 mL/g). Only Tc-99 was considered in the analysis	90	Tc-99
23	Final K_h vault/saltstone to 1×10^{-6} cm/s (1ft/yr)	16	I-129, Se-79, Tc-99
24	Increased precipitation (climate change)	1.1	Se-79, I-129
25	Increased precipitation, pessimistic vault and saltstone degradation	6.5	I-129, Se-79
26	Decreased K_d values by a factor of 10	1.8	I-129, Se-79
27	Pessimistic initial saltstone grout conductivity and pessimistic saltstone grout degradation	0.43	I-129, Se-79
28	Pessimistic initial vault hydraulic conductivity	0.11	Se-79, I-129
29 Ox	100% loss of reducing conditions. Only Tc-99 and U-238 were included in the analysis (hydraulic parameters equivalent to base case values)	3.2	Tc-99

Scenario	Description	Result (mrem/yr)	Radionuclides-Largest Contributors
29 Red	0% loss of reducing conditions. Only Tc-99 and U-238 were included in the analysis (hydraulic parameters equivalent to base case values)	1.6×10^{-13}	Tc-99
30	Oxidizing saltstone and grout combined with increased infiltration and pessimistic vault and saltstone grout degradation. Only Tc-99 and U-238 were included in the analysis	1200	Tc-99
31	Vault/grout 100% saturated, reducing conditions	3.5	Se-79
32	Vault/grout 100% saturated, oxidizing conditions	26	Tc-99, Se-79
33 Ox	Increased infiltration, hydraulic conductivity, diffusivity. 100% oxidizing conditions	34,000	Tc-99, I-129, Se-79
33 Red	Increased infiltration, hydraulic conductivity, diffusivity. 100% reducing conditions	310	I-129, Se-79, Tc-99

Table 9 Description of DOE Sensitivity Analyses

Scenario	Description
1	Base case
2	Optimistic cover degradation - Peak infiltration from 36 cm/yr (14 in/yr) to 18 cm/yr (7 in/yr)
3	Pessimistic cover degradation - Peak infiltration from 36 cm/yr (14 in/yr) to 53 cm/yr (21 in/yr)
4	Optimistic vault degradation - Saturated hydraulic conductivity of the vault changes from 1×10^{-12} to 1×10^{-10} cm/s (1×10^{-6} to 1×10^{-4} ft/yr) over 10,000 years
5	Pessimistic vault degradation - Saturated hydraulic conductivity of the vault changes from 1×10^{-12} to 1×10^{-8} cm/s (1×10^{-6} to 1×10^{-2} ft/yr) over 10,000 years
6	Optimistic initial vault conductivity - Initial saturated hydraulic conductivity of the vault set to 1×10^{-13} cm/s (1×10^{-7} ft/yr)
7	Pessimistic initial vault conductivity - Initial saturated hydraulic conductivity of the vault set to 1×10^{-11} cm/s (1×10^{-5} ft/yr)
8	Optimistic saltstone grout degradation - Saturated hydraulic conductivity of the grout changes from 1×10^{-11} to 1×10^{-10} cm/s (1×10^{-5} to 1×10^{-4} ft/yr) over 10,000 years
9	Pessimistic saltstone grout degradation - Saturated hydraulic conductivity of the grout changes from 1×10^{-11} to 1×10^{-8} cm/s (1×10^{-5} to 1×10^{-2} ft/yr) over 10,000 years
10	Optimistic initial saltstone grout conductivity - Initial saturated hydraulic conductivity of the saltstone grout set to 1×10^{-12} cm/s (1×10^{-6} ft/yr)
11	Pessimistic initial saltstone grout conductivity - Initial saturated hydraulic conductivity of the saltstone grout set to 1×10^{-10} cm/s (1×10^{-4} ft/yr)
12	Pessimistic cover, vault, grout degradation - Peak Infiltration from 14 in/year to 21 in/year. Saturated hydraulic conductivity of the vault goes from 1×10^{-12} to 1×10^{-8} cm/sec (1×10^{-6} to 1×10^{-2} ft/yr) over 10,000 years. Saturated hydraulic conductivity of saltstone grout changes from 1×10^{-11} to 1×10^{-8} cm/sec over 10,000 years (1×10^{-5} to 1×10^{-2} ft/sec)
13	Vault and grout saturated - Relative hydraulic conductivity set to 1 for both the vault and saltstone grout
14	Optimistic vault diffusion - Vault diffusion coefficient set to 1×10^{-9} cm^2/s (3×10^{-5} ft^2/yr)
15	Pessimistic vault diffusion - Vault diffusion coefficient set to 1×10^{-7} cm^2/s
16	Optimistic saltstone grout diffusion - Saltstone diffusion coefficient set to 5×10^{-10} cm^2/s (2×10^{-5} ft^2/yr)
17	Pessimistic saltstone grout diffusion - Saltstone diffusion coefficient set to 5×10^{-8} cm^2/s (2×10^{-3} ft^2/yr)
18	Pessimistic vault/saltstone grout diffusion - Vault diffusion coefficient set to 1×10^{-7} cm^2/s, Saltstone grout diffusion coefficient set to 5×10^{-8} cm^2/s (2×10^{-3} ft^2/yr)
19	Pessimistic K_d values - Partition coefficients for C-14, Se-79, and I-129 set to 0 ml/g
20	Reducing conditions in vault at all times - Tc K_d set to 1000 ml/g. Only Tc-99 was included in the analysis
21	Oxidizing conditions (K_d Tc-99 = 1 ml/g). Only Tc-99 was included in the analysis

Scenario	Description
22	Oxidizing conditions (K_d Tc-99 = 0 ml/g). Only Tc-99 was included in the analysis
23	Final K_h vault/saltstone to 1 x 10^{-6} cm/s (1 ft/yr) - Concrete K_h increases from 1 x 10-12 to1 x 10^{-6} cm/s (1 x 10^{-6} to 1 ft/yr) with a degradation rate constant, α = 3. Saltstone grout K_h increases from 1 x 10^{-11} to 1 x 10^{-6} cm/s (1 x 10^{-5} to 1 ft/yr) with a degradation rate constant, α = 2.5.
24	Increased precipitation (climate change) - Increase the average precipitation used in the base case (i.e., 124 cm/yr [48.8 in/yr]) by 25%.
25	Increased precipitation, pessimistic vault and saltstone degradation - combined scenarios 5, 9, and 24
26	Decrease K_d values in all media by a factor of 10
27	Pessimistic initial saltstone grout conductivity and pessimistic saltstone grout degradation - combined scenarios 9 and 11
28	Pessimistic initial vault hydraulic conductivity - Concrete K_h decreases from 1 x 10^{-8} cm/s to 1 x 10^{-6} cm/s (1 x 10^{-2} to 1 ft/yr) over 10,000 years
29 Ox	100% loss of reducing conditions - Oxidizing K_d values for Tc and U (Bradbury and Sarott, 1995). Only Tc-99 and U-238 were included in the analysis
29 Red	0% loss of reducing conditions (base case redox assumption) - Only Tc-99 and U-238 were included in the analysis. Reducing K_d values for Tc and U (Bradbury and Sarott, 1995)
30	Oxidizing saltstone and grout combined with increased infiltration and pessimistic vault and saltstone grout degradation - combined scenarios 25 and 29. Only Tc-99 and U-238 were included in the analysis
31	Vault/grout 100% saturated, reducing conditions. Assume the vault and saltstone grout exhibit large-scale cracking at a 9.1 m (30 ft) nominal spacing. Assume vault, saltstone grout, and fractures are fully saturated. Implemented by redefining the water retention and relative permeability curves, such that both are 1.0 regardless of suction head.
32	Vault/grout 100% saturated, oxidizing conditions - Same as scenario 31 except for oxidizing conditions
33 Ox	Increased infiltration, hydraulic conductivity, diffusivity. Oxidizing conditions. Infiltration to the vault is set to 25 cm/yr (9.8 in/yr) throughout the simulation and the closure cap drains silted to allow infiltration to go to the saltstone grout. Hydraulic conductivity of the vault and saltstone grout are set to 5 x 10^{-7} cm/s (5 x 10^{-1} ft/yr) throughout the simulation. Effective diffusivity for the vault and saltstone grout are increased by a factor of 10 over the base case. 100% oxidizing conditions at the start of the simulation.
33 Red	Increased infiltration, hydraulic conductivity, diffusivity. Reducing conditions. Same as scenario 33 Ox but for reducing conditions throughout the simulation.

DOE's interpretation of the results in Table 8 are that the Vault 4 special analysis and the results of the sensitivity studies clearly demonstrate that the solidified salt waste from the projected waste stream for Vault 4 can be disposed in Vault 4 and that all applicable performance measures, including the 0.25 mSv/yr (25 mrem/yr) all-pathways public dose limit, will be met. DOE concluded that the results demonstrate that the saltstone disposal system is very robust in that the parameters which are hardest to measure or predict have relatively little impact on the calculated dose. The results of the sensitivity analyses show that the projected dose from the disposal of saltstone is most dependent on the amount of precipitation and resultant infiltration of water through the saltstone grout and the oxidation-reduction condition of the vault and saltstone grout. These parameters are the ones that DOE has a high degree of confidence will perform as described in the special analysis (Cook et al, 2005). DOE has indicated that it is important that the models provide a realistic representation of the physical and chemical processes that will occur within the saltstone disposal vaults and surrounding environment for the next 10,000 years (WSRC, 2005a).

4.2.16 NRC Evaluation – Protection of the Public

DOE has used an all pathways dose assessment to show conformance with the performance objectives established for the public. As indicated in Section 4.2.15, DOE generated results assuming all of the inventory (both existing and projected) would be placed within Vault 4 (0.023 mSv/yr [2.3 mrem/yr]) as well as for the expected Vault 4 inventory (0.00048 mSv/yr [0.048 mrem/yr]). The former case represents an unrealistic scenario that was generated to demonstrate that the performance objectives could be achieved for the disposal system regardless of how the radiological inventory may be distributed in the disposal facility. The latter scenario represents the risk from Vault 4 if DOE were to proceed with the waste treatment and disposal process as currently envisioned. In both cases, the physical and chemical retention capabilities of the system are identical, the only difference is the projected inventory. The base case peak Total Effective Dose Equivalent (TEDE) to a member of the public of 0.023 mSv/yr (2.3 mrem/yr) is well within the performance objective of 0.25 mSv/yr (25 mrem/yr) in 10 CFR Part 61.41 (Protection of General Population from Releases of Radioactivity). There is a substantial difference between the total inventory projected for the SDF and that projected for Vault 4, as discussed below.

Overall, DOE's calculational methodology for performing the PA to evaluate protection of the public is generally consistent with NRC guidance. There are a few areas related to the approach to PA where there is some divergence between NRC and DOE; those areas are discussed in detail in the paragraphs that follow. The selection of the receptor as a future resident that may perform agricultural activities at the site after institutional controls are assumed to cease (100 years) is reasonable considering past regional land use practices prior to development of the SRS. Assuming a member of the public will use water from a well for domestic purposes and that the well is assumed to be located where the maximum concentration of radionuclides in the groundwater are predicted to occur is consistent with NRC's definition of an average member of the critical group. The groundwater pathway calculations were performed in two steps. The stepwise approach to performing the groundwater calculations is reasonable assuming quality controls are implemented to ensure that errors are not introduced. NRC staff did not identify any errors of this type during its review.

The analysis performed by DOE was deterministic. The approach by DOE to evaluate combinations of uncertain parameters with a deterministic analysis is reasonable; however, use of a probabilistic approach can provide many insights into this type of analysis (NRC, 2000b). DOE's model support is limited in a number of areas that the sensitivity analyses identified as impacting system performance, as discussed in detail below.

DOE's initial PA provided limited sensitivity analyses. NRC requested an expansion of the sensitivity analyses in its RAI (NRC, 2005a). Because the PA analyses were deterministic, DOE provided a series of analyses to evaluate the impact of key uncertainties. For complex modeling and long-term projections of system performance, it is essential that uncertainties are evaluated and their impacts factored into decision making (NRC, 2000b). In general, if deterministic modeling is used, it should be reasonably conservative such that a subject matter expert, with minimal interaction with those who performed the assessment, could come to a conclusion that the analysis was conservative. The key uncertainties evaluated by DOE were infiltration through the engineered cap, the hydraulic conductivities of the vault and saltstone and the rate at which they degrade, the effective diffusion coefficients of the saltstone and vault, the impact of fracture flow through cracks in the vault, the saturation state of the vault and saltstone, and the oxidation state of the vault and saltstone. Based on current information, NRC agrees that DOE has evaluated the key uncertainties.

Before discussing NRC's interpretation of DOE's sensitivity analysis results, it is important to address the inventory used in the sensitivity analysis and the potential for multiple sources. Table 10 provides a comparison of the inventory values for the three radionuclides that appear to be most significant to achieving protection of the public through the groundwater pathway. As shown in Table 10, the inventory used by DOE in the sensitivity analysis calculations is projected to be approximately a factor of 3 less than would be expected for the average vault containing salt waste from the tanks. The purpose of the sensitivity analysis was to address uncertainties and provide justification that DOE's salt waste disposal activities could meet the performance objectives; therefore, the use of estimated Vault 4 inventory was inappropriate.

As previously indicated, as many as 14 additional vaults would be needed to dispose of saltstone material. DOE's groundwater modeling to assess the impact from all of the vaults, considering the potential for overlap from plumes of individual vaults at any given potential receptor location, was not completed at the time of the draft waste determination. DOE indicated during the August 17-18, 2005, meeting that their professional opinion was that the impact on groundwater concentrations for the public receptor for the entire SDF would be approximately a factor of 2 larger than for an individual vault (Cook, 2005).

Considering the differences between the projected Vault 4 inventory and the average vault, and the potential for multiple sources (plume overlap from multiple vaults), the sensitivity results presented in Table 8 need to be scaled, at a minimum, to increase by a factor of 6.6 before the significance of the results is interpreted. The 6.6 value is derived by applying a factor of 3.3 for the difference in inventory between the values for projected Vault 4 and the average vault and a factor of 2 to account for overlap of the plumes from multiple disposal vaults. Ideally, these scaled results would have been generated by model calculations. However, in the absence of model results a simple linear scaling should be sufficient for discussion purposes, given the

Table 10 Comparison of Projected Vault 4 Inventory, Total Estimated SDF Inventory, and Average Vault Inventory

Radionuclide	Projected Vault 4 (Ci)	Total SDF (Ci)	Average Vault[1] (Ci)	Ratio[2]
Se-79	1.96	89.4	6.4	3.3
I-129	0.44	18.0	1.3	3.0
Tc-99	716	33,100	2360	3.3

(1) The average vault value is generated by dividing the total SDF inventory by the projected number of vaults (14).
(2) Ratio = Average Vault / Projected Vault 4

current modeling approach to release from the source term (i.e., doses should scale somewhat linearly with inventory when using a K_d-based release model). Table 11 is a list of the sensitivity scenarios provided and the scaled dose result based on the above considerations.

The demonstration that the SDF can meet the performance objective for protection of the public is dependent on the long-term (10,000 year) performance of the engineered closure cap to greatly limit infiltration and the concrete vaults and cementitious saltstone wasteform to limit radionuclide release. The initial analysis performed by DOE had assumed essentially no degradation of the main engineered components of the disposal system (engineered cap, vaults, cementitious wasteform) over 10,000 years in the base case analysis (Cook and Fowler, 1992). In response to the NRC RAI and subsequent meetings with NRC staff, DOE performed analyses to evaluate various degrees of degradation of the engineered components of the disposal system (WSRC, 2005a, b). In providing an interpretation of the sensitivity results, DOE indicated that it considers many of the analyses to represent highly unrealistic conditions. In some instances, NRC staff agrees with the DOE discussion of the likelihood of the sensitivity scenario (i.e., some are clearly unrealistic based on current information and understanding). The major difficulty in interpreting the DOE sensitivity analysis results is the mixture of potentially optimistic and pessimistic conditions or assumptions, and the amount of support provided for the modeling results. Roughly a third of scaled results of the scenarios evaluated in the sensitivity analyses approach or exceed the 0.25 mSv (25 mrem/yr) dose limit, some by a large margin. Of these, 70% represent analyses where 100% oxidizing conditions were assumed at the time of facility closure. The following bullets provide a synthesis of NRC's assessment of the realism of DOE's sensitivity scenarios.

- NRC agrees with DOE that the assumption of 100% oxidation of the waste at year zero is an unrealistic assumption. Scaling of the sensitivity results to the amount of oxidation expected by DOE over 10,000 years (3 to 8%) is confounded by the fact that this amount of oxidation is projected for the next 10,000 years (i.e. it will not occur instantaneously) and will likely occur at exposed surfaces rather than throughout the bulk of the material. Assuming 0% oxidation (DOE base case) is clearly non-conservative while assuming 100% oxidation is unrealistic.

Table 11 Scaled Results for the Sensitivity Scenarios

Scenario	Description	Result (mrem/yr)
1	Base case	0.32
2	Optimistic cover degradation	0.13
3	Pessimistic cover degradation	1.8
4	Optimistic vault degradation	0.20
5	Pessimistic vault degradation	0.34
6	Optimistic initial vault conductivity	0.21
7	Pessimistic initial vault conductivity	0.34
8	Optimistic saltstone grout degradation	0.24
9	Pessimistic saltstone grout degradation	1.5
10	Optimistic initial saltstone grout conductivity	0.25
11	Pessimistic initial saltstone grout conductivity	1.6
12	Pessimistic cover, vault, grout degradation	26
13	Vault and grout saturated	1.2
14	Optimistic vault diffusion	0.23
15	Pessimistic vault diffusion	1.1
16	Optimistic saltstone grout diffusion	0.26
17	Pessimistic saltstone grout diffusion	0.42
18	Pessimistic vault/saltstone grout diffusion	4.4
19	Pessimistic K_d values	240
20	Reducing conditions in vault at all times. Only Tc-99 was included in the analysis	1.0×10^{-12}
21	Oxidizing conditions (K_d Tc-99 = 1 mL/g). Only Tc-99 was included in the analysis	21
22	Oxidizing conditions (K_d Tc-99 = 0 mL/g). Only Tc-99 was included in the analysis	590
23	Final K_h vault/saltstone to 1×10^{-6} cm/s (1 ft/yr)	110
24	Increased precipitation (climate change)	7.3
25	Increased precipitation, pessimistic vault and saltstone degradation	43
26	Decreased K_d values by a factor of 10	12
27	Pessimistic initial saltstone grout conductivity and pessimistic saltstone grout degradation	2.8
28	Pessimistic initial vault hydraulic conductivity	0.73
29 Ox	100% loss of reducing conditions. Only Tc-99 and U-238 were included in the analysis	21

Scenario	Description	Result (mrem/yr)
29 Red	0% loss of reducing conditions (base case redox assumption). Only Tc-99 and U-238 were included in the analysis	1.0×10^{-12}
30	Oxidizing saltstone and grout combined with increased infiltration and pessimistic vault and saltstone grout degradation. Only Tc-99 and U-238 were included in the analysis	7900
31	Vault/grout 100% saturated, reducing conditions	23
32	Vault/grout 100% saturated, oxidizing conditions	170
33 Ox	Increased infiltration, hydraulic conductivity, diffusivity. 100% oxidizing conditions.	220,000
33 Red	Increased infiltration, hydraulic conductivity, diffusivity. 100% reducing conditions.	2000

- The amount and type of degradation experienced by the saltstone wasteform over 10,000 years was evaluated in a number of analyses. The amount of degradation assigned by DOE was based on professional judgment, informed by qualitative consideration of degradation mechanisms and other factors. In DOE's base case, the hydraulic conductivity of the saltstone was assumed to change from 1×10^{-11} cm/s to 1×10^{-9} cm/s (1×10^{-5} to 1×10^{-3} ft/yr), the effective diffusivity remained unchanged throughout the analysis at 5×10^{-9} cm^2/sec (2×10^{-4} ft^2/yr), and the wasteform experienced no oxidation. In scenario 25, the hydraulic conductivity of the saltstone and vault were assumed to degrade to 1×10^{-8} cm/s (1×10^{-2} ft/yr) and the precipitation was increased by 25%. Even with no oxidation of the wasteform, the scaled result is 0.43 mSv/yr (43 mrem/yr), which would exceed the 0.25 mSv/yr (25 mrem/yr) performance objective. This result demonstrates that proper design of the engineered barriers and careful implementation of the design is essential to meeting the performance objective for protection of the public. In addition, the modeling results or assumptions for the degradation of the saltstone, vaults, and engineered cap must be supported by adequate information.

- The initial values for the hydraulic conductivity of the saltstone and vaults are based on very limited measurements of small-scale samples. The thermal and mechanical conditions experienced by the field-scale saltstone wasteform during curing may be significantly different than the conditions experienced by the samples that have been tested. While the sensitivity analyses demonstrate that the model results are not very sensitive to the initial values of hydraulic conductivities, testing performed on earlier saltstone formulations exhibit much larger hydraulic conductivities and possibly indicate temporal variability associated with curing time (Malek et al., 1985).

- Fracture flow was assessed in scenario 31 for no oxidation of the waste and scenario 32 for complete oxidation of the waste. Fractures extending through the vaults were introduced with a 9 m (30 ft) spacing parallel to the short axis of the vault based on analysis performed of settlement and seismic effects (Peregoy, 2003). The fractures, saltstone, and vaults were assumed to be saturated. The scaled result for scenario 31 and 32 were 0.23 mSv/yr

(23 mrem/yr) and 1.7 mSv/yr (170 mrem/yr), respectively. Assuming 100% saturation of the system and complete oxidation of the waste is not realistic.

- Some degree of natural (non-human induced) climate change can be expected to occur over the next 10,000 years at the SRS. Scenario 24 demonstrates that the current model results are sensitive to estimates of infiltration through the engineered cap. Assuming a 25% increase in precipitation at year zero is unrealistic. Assuming no change in the climate (considering the model sensitivity) over the next 10,000 years (DOE base case) is non-conservative.

- Relative permeability values developed for the saltstone and vaults were determined to have been obtained from experimental results that are likely in error. Sensitivity scenario 13 assumed that the vault and grout were saturated throughout the analysis. This assumption is unrealistic. The impacts of using the questionable data was bound by a factor of 4.

Scenario 33 was completed at the request of NRC. NRC staff wanted to understand what the conservative upper bound for the risk to the public from the disposal of saltstone may be. The scenario combined an infiltration rate of 25 cm/yr through the upper GCL, an assumption of complete siltation of the lower drainage layer of the engineered cap, an assumed value for hydraulic conductivity of the vault and saltstone of 5×10^{-7} cm/s (5×10^{-1} ft/yr), an increase of effective diffusivity of the vault and saltstone by a factor of 10, and either completely reducing or completely oxidizing conditions. DOE describes this scenario as “extremely conservative in that it represents a disposal system that has no closure cap and no vault and in which the saltstone grout had properties similar to SRS sandy clay soil” (WSRC, 2005a). The NRC staff agrees that it is overly conservative to assume that complete oxidation of the saltstone will occur, but does not believe that this scenario is extremely conservative if it is assumed there is no oxidation. NRC staff interprets this analysis to represent a disposal system that has an engineered cap that has not performed as designed, resulting in an infiltration rate consistent with observations of natural recharge rates at the site. The hydraulic conductivity and effective diffusivity values used for the vault and saltstone do not represent a system with no vault and grout having properties of SRS sandy clay soil, but instead represent degraded cementitious materials with properties that are roughly 100 times better than a SRS sandy clay soil and closer to a controlled compacted kaolin or a natural clay deposit (Phifer and Collard, 2003). The analysis demonstrates that DOE must appropriately design the engineered systems, and develop the model support to confirm that it will perform as designed, or, based on current modeling, it is questionable that the performance objective for protection of the public can be met. The analysis also demonstrates that more realistic modeling of infiltration, water contact with the waste, waste oxidation, and radionuclide release in an unsaturated and potentially fractured system is needed, because conservative modeling does not yield acceptable results.

In (WSRC, 2005a), DOE provided normalized sensitivity as a metric to interpret the sensitivity analysis results. Using the normalized sensitivity measure eliminated the effects of units and the absolute magnitude of each parameter. However, the challenge with using normalized sensitivity is that the results can be influenced by the range of the parameter that is evaluated. For example, the dose response to a parameter may not be linear. Therefore if an arbitrary large range is selected, the sensitivity measure can be in effect diluted. Sensitivity should be considered with respect to the level of knowledge of the range of the parameter being investigated to avoid this impact if possible.

The deterministic DOE base case result does not have adequate model support and is not conservative. Considering the uncertainty in many key parameters, it should not be used as the basis for demonstrating that there is reasonable assurance that the performance objective for protection of the public can be met or for developing inventory limits. A revised base case should be based on the projected average vault inventory for salt waste from the tanks and the orientation of multiple disposal vaults, include the expected magnitude and timing of climate change from the natural cycling of climates, include the expected magnitude and rate of oxidation of waste, consider liquid and gas flow in fractures (that may develop), and account for the questionable moisture characteristic curve information for concrete and saltstone that was used in the previous analysis.

Model support is essential for deterministic modeling that is not clearly conservative because uncertainties are not represented in a deterministic analysis. Model support is information that provides confidence that the numerical model results are adequately representing the behavior of the actual system. Model support is commonly called model validation in the technical literature, but NRC recognizes that some aspects of PA are not amenable to scientific model validation (e.g., the long time frames) (NRC, 2000b). Model or software verification is a process to ensure that the numerical calculations are correct, but it does not ensure that the model results are appropriate for the given application. It does appear that DOE has performed adequate software verification for the main products it used to complete the PA (e.g., PORFLOW, HELP, internally developed intruder analysis software). NRC requested additional model support for key areas of DOE's PA in the RAI (NRC, 2005a). DOE stated that the information provided in (WSRC, 2005b) encompasses all currently available field data providing model support for the simulation results. A path forward that outlined how DOE intended to develop adequate model support was discussed in the August 17, 2005, meeting; however, a path forward was not provided in the written response to the action items (WSRC, 2005a).

The following example is intended to illustrate the types of information that could be developed to provide adequate model support. One of the key elements of DOE's PA is the reduction of Tc-99 in the wasteform by the addition of slag. As previously discussed, the sensitivity analyses demonstrate quite clearly that the rate and extent of oxidation of the wasteform is a key factor in meeting the protection of the public performance objective. DOE has performed basic research to evaluate whether the slag would result in Tc-99 being contained in a reduced form, and installed field-scale saltstone lysimeter tests with and without slag (Cook and Fowler, 1992). The slag-based lysimeters showed a much lower release rate of Tc-99 compared to the lysimeters without slag. At the request of NRC, DOE provided a comparison of a PORFLOW simulation with analytical data, that showed an underprediction by the model of roughly a factor of 10 (WSRC, 2005a). Figure 7 provides the comparison of the PORFLOW and slag lysimeter data. DOE explained the underprediction was the result of the analytical measurements being performed for gross non-volatile beta-gamma emitters, and therefore the results show essentially background. However, the data shows an increasing trend starting around day 250 that is consistent with the PORFLOW simulation, therefore it is unclear that the analytical data represents background. In addition to providing information on field-scale release rates, the slag lysimeters may contain valuable information about the rate of oxidation of the saltstone. Currently, DOE's estimates for the amount of oxidation of the saltstone over 10,000 years are based primarily on numerical modeling results. It may be possible to exhume and characterize a saltstone lysimeter. The depth of the penetration of the oxidation front should be able to be estimated and it would provide excellent model support for a key element of DOE's PA.

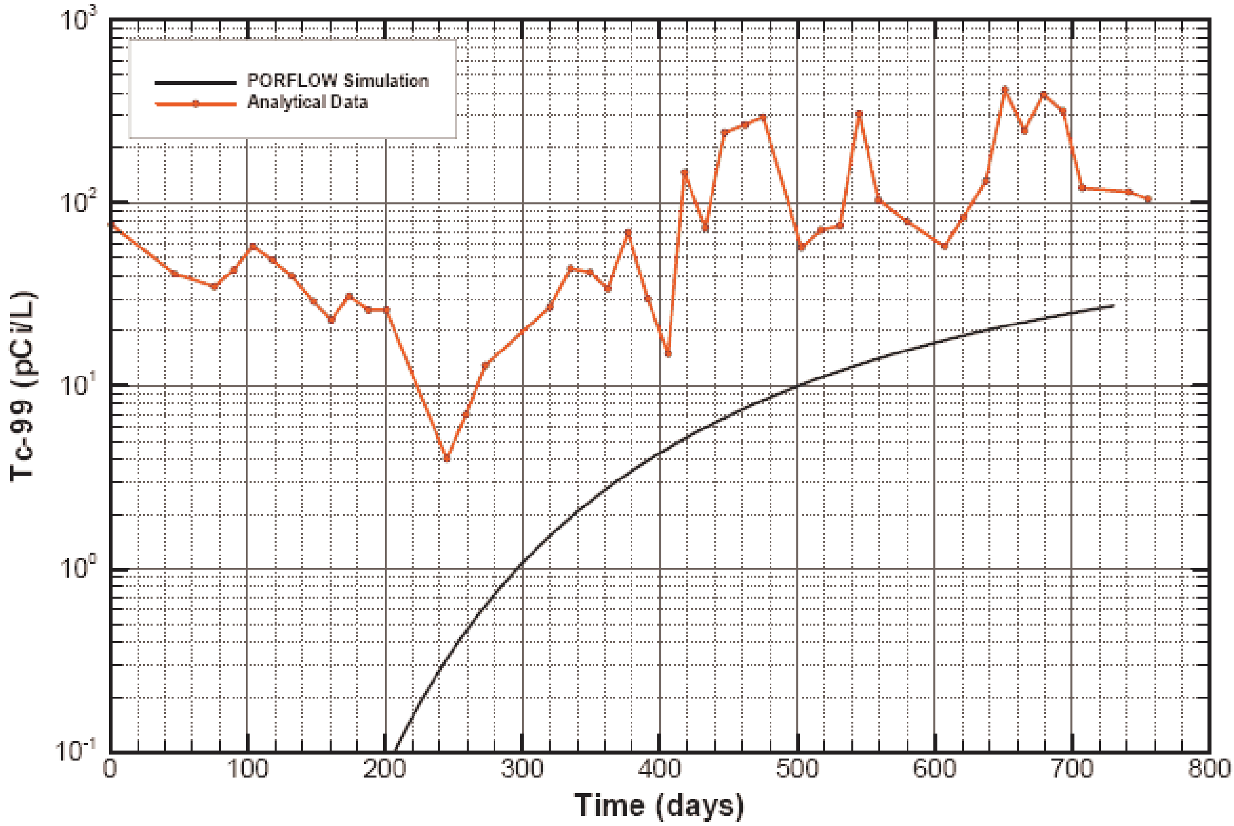

Figure 7 Comparison of PORFLOW and Slag Lysimeter Data

In order to better risk-inform the staff's review, the NRC staff developed a PA model applicable to the SDF using the software platform GoldSim (Kossik, 2005). The model was used to evaluate sensitivity of the PA results and to corroborate, in a general sense, the DOE-SRS calculational results. Utilization of the model by the staff allowed a more focused review on those technical aspects of the problem more likely to influence the risks. With similar input values for parameters and assumptions, the output from the simulations in NRC's independent analysis was in general agreement with DOE's PA.

NRC staff concludes that there is reasonable assurance that the performance objectives in 10 CFR 61.41 can be met, including the provision of as low as reasonably achievable (ALARA) releases of radioactivity to the general environment, if the following conditions are satisfied: (1) DOE will perform more realistic modeling of waste oxidation and moisture movement with results that demonstrate release rates are acceptable, (2) DOE will perform confirmatory activities that demonstrate the hydraulic conductivity of the saltstone and vault will not degrade to values larger than 1 x 10^{-7} cm/s (1 x 10^{-1} ft/yr), and (3) DOE will enhance model support for the areas discussed above. The ALARA provision is not part of the PA calculation, as the PA is the means to generate results to compare to performance objectives. Through demonstration of Criterion Two (the waste has had highly radioactive radionuclides removed to the maximum extent practical), DOE can satisfy the intent of the ALARA provision to maintain releases of radioactivity to the environment as low as reasonably achievable.

4.2.17 Protection of Intruders

In estimating doses to inadvertent intruders to the site after the 100 year period of institutional control, it is assumed that individuals could establish a residence and that the intruders have no previous knowledge of waste disposal activities at the site. For assessing whether the performance objective of 10 CFR 61.42 could be met, DOE's analysis used the projected inventory for Vault 4 (Rosenberger, et al., 2005). DOE indicated that the Vault 4 inventory was expected to bound the inventory for other vaults because the dominant radionuclide for the intruder scenarios was Cs-137 and the DDA waste stream, which has the highest Cs-137 concentration, will be located in Vault 4. Direct intrusion into the SDF facility was assumed to occur under one of three scenarios: agriculture, resident, and post-drilling. Acute and chronic exposures were considered. However, only chronic exposures were evaluated for the SDF because of the longer exposure times associated with chronic scenarios.

Figure 8 is a visualization of the resident intruder pathway. The resident intruder scenario is assumed to be a credible occurrence at any time after the active institutional controls are relinquished. It is expected that institutional controls of various form (active, passive) will be maintained at the site longer than the 100 years assumed in the analysis. Surveillance and maintenance will be used during the period of institutional controls to repair any degradation to the engineered cap. In the resident intruder scenario, the intruder is assumed to excavate a foundation for a home to a maximum depth of 3 m (10 ft). Over time erosion will lower the ground surface until the erosion barrier layer of the engineered cap becomes exposed. Thus the amount of shielding will decrease over time 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 after closure. The inadvertent intruder analysis was performed by an automated code developed by Savannah River National Laboratory (SRNL) (Koffman, 2004).

The erosion barrier layer has been designed to maintain a 3 m (10 ft) thickness of soil over the top of the vaults for 10,000 years (Cook, et al., 2005). The erosion barrier is to be constructed of durable material sized to remain in place during the maximum rainfall event expected over the 10,000 year design (Phifer and Nelson, 2003). Sections 4.2.4 and 4.2.5 provide a detailed discussion of the erosion control design. The agricultural scenario is not possible if the erosion barrier performs as designed, because the soil layer prevents waste from being exhumed.

The reinforced concrete vault roof and saltstone waste form are assumed to present a barrier that would discourage drilling activities, and therefore drilling is not considered a plausible scenario. The thick cover system provides protection from physical weathering. Initially and for many years after construction, the vaults and wasteform will present a dramatically more difficult media through which to drill. The concrete and saltstone grout will undergo chemical degradation over time which will slowly alter the properties of the materials. However it is expected that even altered concrete and saltstone grout will present a sharp contrast to the native sands and clays. The well-driller intruder is not deemed credible for the SDF based on the long-term physical properties of the materials and the regional practices for well development.

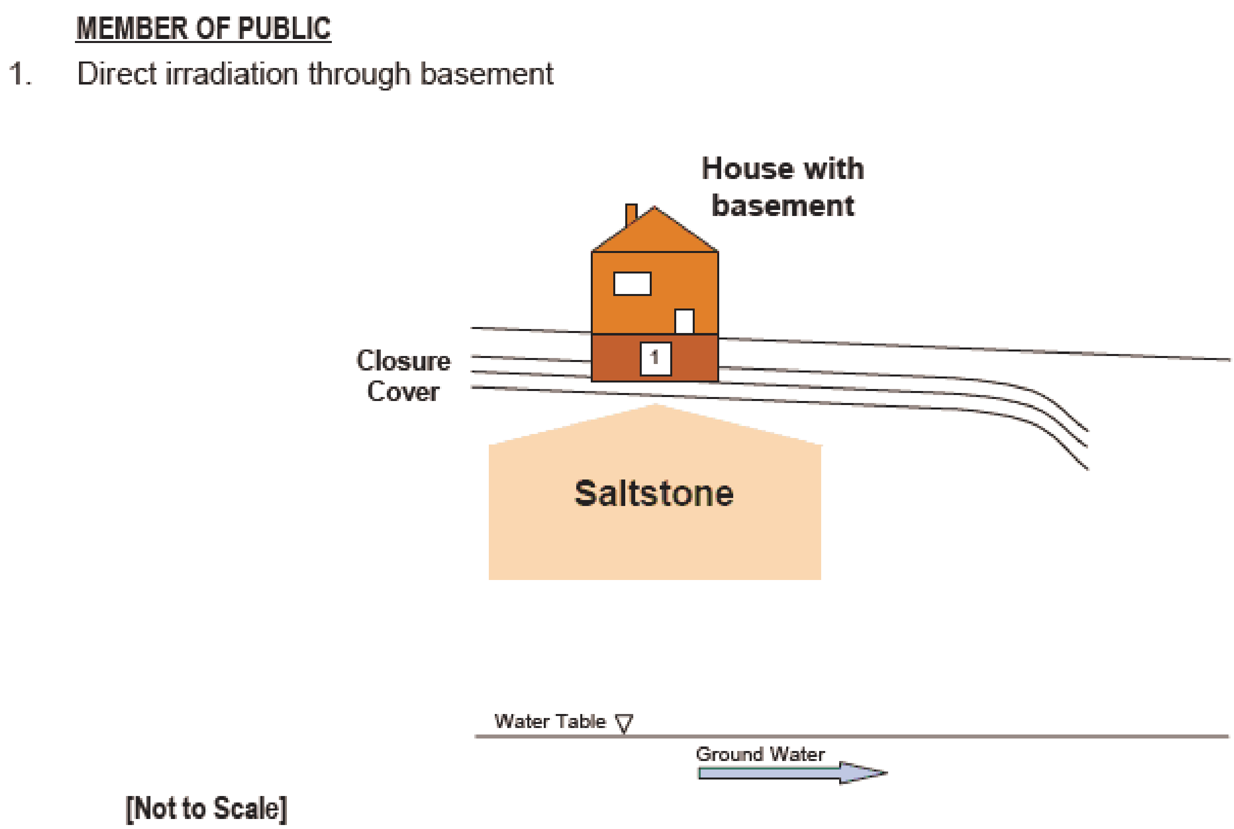

Figure 8 Resident Intruder Scenario

The projected Vault 4 inventory and the inventory limit for each radionuclide was used to estimate the dose contribution regardless of the time of occurrence. The dose contributions were then summed to estimate the total dose, yielding 0.21 mSv/yr (21.7 mrem/yr), which is approximately 4% of the 5 mSv/yr (500 mrem/yr) performance objective. The four largest contributing radionuclides were Cs-137 (92%), Sn-126 (3.7%), Al-26 (2.9%), and U-234 (0.7%). A geometrical reduction factor was applied to account for the probability of placing a house on the area between the vaults, thereby reducing the exposure to the "average" intruder (Cook et al., 2005). Without taking into account the geometrical reduction factor, the resident intruder dose would be 0.36 mSv/yr (36 mrem/yr).

The agricultural and drilling scenarios were assessed by DOE as part of sensitivity studies in response to the NRC RAI, in part to address NRC concerns regarding the long-term performance of the erosion control barrier (WSRC, 2005a). For the drilling scenario, it was assumed at 1,000 years after closure that a well is drilled through the disposal vault. The subsurface material exhumed during the drilling included some of the saltstone waste. The exhumed material was assumed to be mixed with soil in a garden and the intruder is exposed to the waste through a variety of pathways (e.g., direct radiation from contaminated soil, ingesting contaminated food from the garden, direct ingestion of contaminated soil, and inhalation of radionuclides attached to soil particles). A dilution factor of 0.02 was applied to account for mixture of the exhumed waste in the vegetable garden (Rosenberger et al., 2005). The sum of fractions derived for this scenario was 0.31 resulting in an estimated post-drilling intruder dose of 0.31 mSv/yr (31 mrem/yr). For the agricultural scenario, it was assumed that the erosion barrier did not function as designed, such that it eroded at the same rate as the native soil. This allowed the agricultural scenario to occur after a sufficient depth of the closure cover had eroded away. The intruder was assumed to build a home directly on top of the disposal units, and the foundation was assumed to extend into the disposal units. Waste exhumed in this scenario is assumed to be indistinguishable from native soil. Exposure pathways are similar to the post drilling scenario. A dilution factor of 0.2 was applied to account for mixture of the exhumed waste with native materials (Rosenberger et al., 2005). Based on an earlier estimate of projected Vault 4 inventory, the sum of fractions for this scenario was 1.49 (based on 1 mSv/yr [100 mrem/yr]) which is equivalent to 1.49 mSv/yr (149 mrem/yr) and less than the 5 mSv/yr (500 mrem/yr) NRC performance objective for inadvertent intruders (Cook et. al, 2005).

4.2.18 NRC Evaluation – Protection of Intruders

It is difficult to predict future actions of humans over hundreds to thousands of years. The intruder scenarios are a regulatory construct designed to ensure protection of the public from unlikely events while eliminating speculation about future human activities. The intruder analyses assume that humans will partake in normal land use activities consistent with regional practices when the active institutional controls are no longer enforced (i.e., disrupt the waste at 100 years with no consideration of the likelihood of occurrence). DOE anticipates they will maintain active control of the site longer than 100 years. A numerical performance objective is not provided in 10 CFR Part 61.42, however a dose limit of 5 mSv (500 mrem) per year was described in the Draft Environmental Impact Statement for 10 CFR Part 61 for development of waste classification requirements and is applied here for intruder scenarios (NRC, 1981).

DOE considered reasonable intruder scenarios to evaluate protection of inadvertent intruders and demonstrate that the performance objectives in §61.42 (Protection of Individuals from Inadvertent Intrusion) could be achieved. Intruder protection is based on chronic exposures associated with the resident intruder scenario. In the resident intruder scenario, a home is excavated above the disposal units and the receptors are exposed to direct radiation that has been attenuated by the intact saltstone vault roof, soil, and the foundation of the house. Ingestion pathways are eliminated from consideration by the presence of at least 3 m of soil over the disposal vaults by the engineered closure cap (see Section 4.2.5). Intruder doses for the resident scenario are calculated to be less than the 5 mSv/yr (500 mrem/yr) performance objective.

DOE indicated that "the use of the average concentrations of radionuclides in a disposal vault, rather than the maximum concentrations at any location in a vault, is appropriate when an inadvertent intruder would access a vault at random locations" (Cook et al., 2002). From a risk perspective, the statement is correct. However, the information provided in (DOE, 2005) shows that each waste stream may in fact be different classes of waste (Class A, B, or C). Thus the risk from each type of vault should be provided, unless the waste streams are going to be mixed prior to emplacement in the vaults. The reduced likelihood of the scenario occurring is already accounted for in the application of a 5 mSv/yr (500 mrem/yr) limit to the intruder scenarios as compared to the application of a 0.25 mSv/yr (25 mrem/yr) limit to the member of the public, which is the nominal scenario. Use of the average concentration is not appropriately protective if the volume of more highly-concentrated waste would fill an area that is consistent with the exposure scenario. If the volume of waste is considerably smaller than the area used in the exposure scenario, then averaging would be appropriate. DOE applied a reduction factor of 0.6 to account for the likelihood that the residence may be located in the area between vaults on the disposal facility. As indicated with respect to waste concentrations, the likelihood of the scenario occurring is accounted for in application of the higher limit. Because the resident intruder doses are dominated by Cs-137 and DOE used the estimated inventory for Vault 4, use of the average concentration for Vault 4 is protective for the resident intruder. The Vault 4 inventory is projected to contain 1.2 x 10^6 Ci of the 3 x 10^6 Ci total Cs-137 to be disposed of in the SDF. Elimination of the reduction factor would increase the estimated resident intruder dose from 0.217 mSv/yr (21.7 mrem/yr) to 0.36 mSv/yr (36 mrem/yr), which is well within the 5 mSv/yr (500 mrem/yr) performance objective. This type of reduction factor should not be applied in future waste determinations.

The magnitude of the intruder doses are strongly influenced by the amount of shielding provided by the vaults and closure cap. For example, in an earlier analysis DOE estimated that a reduction in the amount of shielding from 0.75 m to 0.5 m at year 100 would increase doses by about a factor of 13 (Cook et al., 2002). It was also estimated that as a result of much larger Cs-137 inventory than projected in 1992, the intruder doses would exceed the performance objective by a large margin at 100 years with inadequate shielding (Cook et al, 2002). DOE has designed a thick, persistent closure cap to ensure that adequate shielding will be provided. Because Cs-137 has a relatively short half life (30 years) compared to the analysis time frame (10,000 years), the risk to the resident intruder decreases rapidly and would meet acceptable values on the order of a few hundred years after disposal facility closure even with reduced shielding (e.g., failure of the engineered cap). There is a higher degree of confidence that an erosion control barrier can be designed and will perform acceptably for hundreds of years compared to tens of thousands of years.

DOE assessed the agricultural and drilling scenarios as part of sensitivity studies in response to the NRC RAI, in part to address NRC concerns regarding the long-term performance of the erosion control barrier (WSRC, 2005a). The sum of fractions derived for the drilling scenario was 0.31 resulting in an estimated post-drilling intruder dose of 0.31 mSv/yr (31 mrem/yr). The dose for the agricultural scenario was 1.49 mSv/yr (149 mrem/yr), which is less than the 5 mSv//yr (500 mrem/yr) NRC performance objective for inadvertent intruders. The results for the post-drilling and agricultural scenario are based on estimates for projected Vault 4 inventory that can be considerably less than than the average inventory expected for each of the vaults. Cook et al., 2002 used an inventory of 2.65 Ci for Sn-126 and 98.2 Ci for Tc-99. However, the average inventory of the vaults (assuming 14 vaults) would be approximately 32.1 Ci for Sn-126 and 2357 Ci for Tc-99 (d'Entremont and Drumm, 2005). Therefore, scaling the agricultural intruder doses to account for estimated average inventory of the vaults would result in approximately 6.31 mSv/yr (630 mrem/yr) for Sn-126 and 2.21 mSv/yr (220 mrem/yr) for Tc-99. Based on current inventory estimates and if the erosion control barrier was to not perform as designed, these results indicate that the long-term intruder doses would likely approach or exceed the performance objective for protection of inadvertent intruders. Given the robust erosion control design, the likelihood that the agricultural intruder scenario is applicable is significantly reduced, though not completely eliminated given the uncertainty in projecting the performance of engineered systems over very long periods of time. In the event that the erosion control design does not function properly, the dose is not excessive. The agricultural intruder dose would exceed the value Sn-126 due to the contribution from other radionuclides. However, the results for Tc-99 and Sn-126 can not be directly summed because of differences in the timing of when the peaks would be expected to occur.

DOE's approach to estimating intruder doses is to sum the peak contribution to dose by radionuclide regardless of the time of the peak dose. This approach should yield a conservative estimate of dose because the time of the peak dose from different radionuclides would be expected to be different. DOE could consider summing the contribution from each radionuclide at each point in time and reporting the peak of the sums. The approach being used by DOE may be overly conservative under certain circumstances.

It is expected that the maximum acute dose a hypothetical resident intruder would receive is 0.36 mSv/yr (36 mrem/yr), which demonstrates reasonable assurance that the performance objective for protection of intruders can be met. Considering the physical properties of the vaults and saltstone, the NRC agrees that the post-drilling scenario would be unlikely, especially for the next 1,000 years. If a drilling event were to occur after 1,000 years, the performance objective could still be met. The erosion control design should eliminate the occurrence of the agricultural scenario in the near-term, and greatly reduce the likelihood of occurrence in the long-term.

4.2.19 Protection of Individuals During Operations

The performance objective in 10 CFR 61.43 cross-references "the standards for radiation protection in Part 20". DOE's approach to demonstrating protection of individuals during operations (10 CFR 61.43) is to cross-walk the relevant DOE regulation or limit with that provided in 10 CFR Part 20 and demonstrate that the DOE regulation provides an equivalent level of protection. The cross-referenced "standards for radiation protection" in 10 CFR Part 20 (USNRC, 2005) that are considered in detail 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). 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.

A number of measures will ensure that exposure of individuals during operations are maintained ALARA. These include: (1) a documented Radiation Protection Program (RPP), (2) a Documented Safety Analysis (DSA), (3) design of the SDF and SPF, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, and dosimetry (Rosenberger et al., 2005). The discussion that follows was provided in Rosenberger et al., 2005.

A DSA has been approved by DOE for operation of the SPF and SDF in accordance with 10 CFR Part 830 (WSRC, 2004a). As the first step in the development of the DSA, a formal Consolidated Hazards Analysis (CHA) (WSRC, 2004b) was performed 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 basic 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 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 the 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 then analyzed the hazards that were identified in the CHA that could impact facility workers during normal operations and accident conditions.

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. Dose rate calculations are being performed to determine the shielding requirements for the facility. While the upgraded design is not yet complete, based on the current SPF and SDF design, DOE estimates 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 50 mSv per year (5 rem per year) during both Interim Salt Processing and SWPF operation (WSRC, 2005b).

The effectiveness of the radiation protection programs at the Savannah River Site has been demonstrated by past occupational exposure results. Since 1998, the highest dose received by an SRS worker has been 18.08 mSv (1808 mrem) TEDE compared to a DOE Administrative Control Limit of 20.0 mSv per year (2000 mrem per year) and the 10 CFR Part 835 limit of 50.0 mSv per year (5000 mrem per year). There has been close to zero total exposure for the SPF and SDF workforce. The total dose received by workers at the SPF and SDF since 1998 is 0.35 mSv (35 mrem) (Rosenberger et al., 2005).

The air pathway is the predominate pathway for doses to the public from SDF operations. Doses from the air pathway to members of the public have been, and are expected to continue to be, well below the 1.0 mSv (0.1 rem) annual limit.

4.2.20 NRC Evaluation – Protection of Individuals During Operations

DOE has provided adequate information that individuals will be protected during operations. DOE provided a detailed cross-walk of the relevant DOE regulations to those provided in 10 CFR Part 20, which is referenced in the 10 CFR 61.43 performance objective. NRC agrees that an equivalent level of protection is provided by the relevant DOE regulations or limits to the requirements found 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). In addition, a number of measures are applied to ensure that exposure of individuals are maintained as low as reasonably achievable including: (1) a documented Radiation Protection Program (RPP), (2) a Documented Safety Analysis (DSA), (3) design of the SDF and SPF, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, and dosimetry.

In general, the activities at the SPF and SDF involve inert materials and common, low-temperature, low-energy industrial operations. The public will be located a significant distance (several kilometers) from the facilities during operations and active security is maintained to prevent inadvertent access to the site. The NRC agrees with DOE that the risk to the public during operations should be minimal, and the relevant regulatory limits can be achieved.

The existing DSA for the SPF does not address the organic material in the waste streams resulting from MCU and SWPF, or the processing of the organic salt waste stream from Tank 48. To prepare for future activities, a hazard analysis and DSA revision process will be completed to provide the safety basis for the activities. WSRC is in the process of obtaining information to understand the potential hazards associated with the processing of organic materials in the SPF and SDF. The information will be considered when the CHA process is applied to the future waste streams.

4.2.21 Site Stability

The saltstone facility is located approximately 10 kilometers (6.2 mi) from the nearest site boundary. The site was selected for low-level waste disposal because of its location on a well-drained topographic high. The probable maximum flood for the nearby surface water body (Upper Three Runs) is 53 m (175 ft) above mean sea level, which is substantially below the planned maximum depth of the SDF vaults (WSRC, 2004a). The SDF has been sited in an area of relatively low seismic activity. The largest known earthquake to affect SRS was the Charleston earthquake of 1886. The epicenter was approximately 144 km (90 mi) from SRS and had a magnitude of 6.6 on the Richter Scale. It has been estimated that an earthquake of this magnitude would result in a peak ground acceleration of 0.10g at SRS, whereas a seismic evaluation showed that the soils beneath Z-Area are not susceptible to significant liquification for earthquakes having a peak ground acceleration less than or equal to 0.17g (URS/Blume, 1982), (McHood, 2002).

Two features provide the primary stability for the SDF: (1) the wasteform and engineered vaults and (2) the engineered cap and erosion control design. The wasteform and vaults are cementitious materials designed to provide a solid monolith with little void space. There is expected to be a small amount of shrinkage of the saltstone resulting in a small gap (< 1 cm) between the saltstone and the walls of the disposal vaults. The use of cement and concrete in the wasteform and disposal vault will eliminate differential settlement commonly observed in low-level waste disposal. Although cracking of the vaults and wasteform may be expected over long periods of time, significant structural collapse is not anticipated. It is currently envisioned that a thick, multi-layer engineered cap will be placed over the disposal vaults at closure. An erosion design with an erosion control barrier will be used to ensure that at least 3 m (10 ft) of soil will be maintained over the disposal vaults. There are no known biotic disruptive processes that would significantly impact the disposal system at the SDF site.

The SDF is an operational low-level waste disposal facility for which additional disposal vaults will be constructed for disposal of the tank salt waste. Because final closure will not occur until 2019, a final closure plan has not yet been developed. Actions that are contemplated for final closure have been given in Phifer and Nelson, 2003 and in Cook et al., 2005. The 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. To further ensure long-term stability of the disposal site, the land in Z-Area will remain under the ownership of the Federal Government. The three counties making up the SRS have zoning restrictions that prevent the purchase of property or the approval of building permits at SRS. Residential use of this land will be prohibited via continued land use leasing restrictions.

4.2.22 NRC Evaluation – Site Stability

DOE's plans to place the saltstone grout wasteform in concrete vaults eventually covered by a thick, engineered cap specifically designed for erosion control appear sufficient to indicate that there is reasonable assurance that the performance objective of §61.44 (Stability of the Disposal Site After Closure) can be met. The wasteform and vaults will likely contain minimal void space, therefore differential settlement and the associated negative effects on waste isolation would be eliminated. The SDF is not located in the flood plain of nearby surface water bodies and seismic impacts are expected to be limited to cracking of the disposal units along construction joints.

Erosion is the main disruptive process that may influence the stability of the disposal site. As indicated in Section 4.2.5.1, DOE has provided sufficient information to conclude that the erosion protection design is adequate to provide reasonable assurance of long-term stability of the closure cap for erosion control purposes.

4.3 NRC Review and Conclusions [Criterion Three (A)]

The NRC staff concludes that there is reasonable assurance that Criterion Three (A) will be met for disposal of salt waste in the saltstone disposal facility. As a result of reviewing the DOE analysis and supporting technical information, staff concludes that the saltstone disposal facility, saltstone wasteform, and the engineered closure cap must be appropriately designed and the designs appropriately implemented in order to satisfy the performance objectives of 10 CFR 61 Subpart C, in particular 10 CFR 61.41. As with any disposal action involving a persistent hazard, there is significant uncertainty in the projected performance of engineered systems over extended periods of time. DOE has taken or will be taking actions to mitigate those uncertainties.

The NRC staff's conclusion is based on a number of assumptions about technical aspects of the disposal facility and wasteform. Many of these assumptions relate to the mechanisms and rates of degradation of the disposal facility's engineered features. NRC staff conclusions are conditional on verification of the assumptions. For the SDF, demonstration that 10 CFR 61.41, "Protection of the General Population from Releases of Radioactivity" can be met appears to pose the biggest challenge.

As discussed in previous sections of this TER, maintaining Tc-99 in the wasteform in a reduced form to limit its release to the groundwater pathway is the key factor in achieving the performance objective. The DOE wasteform formulation contains blast furnace slag in an attempt to immobilize technetium. DOE field lysimeter experiments have shown that the release of technetium from wasteform samples that contain slag is significantly slower than the release of technetium from samples that do not contain slag. NRC staff assumed that more realistic modeling of waste oxidation and release of technetium from an oxidized layer of waste will result in predicted doses that are significantly lower than those projected in the DOE sensitivity analysis for 100% oxidation of the waste (see Section 4.2.9). DOE currently estimates that 3 to 8% of the wasteform will oxidize during the 10,000 year performance period; therefore the fractional rate of wasteform oxidation is likely to be small enough to allow 10 CFR 61.41 to be satisfied.

Limiting the amount of water flow through the wasteform is important to achieving 10 CFR 61.41. NRC staff have assumed that the wasteform will not have a bulk degraded hydraulic conductivity greater than 1×10^{-7} cm/s. Bulk degradation of the wasteform can occur from a variety of mechanisms, as discussed in Section 4.2.9. While the vault contains rebar and other embedded carbon steel components, the saltstone wasteform is free of these materials and therefore would not be expected to experience significant changes in bulk hydraulic properties from reinforcement corrosion. The thick engineered cover will protect the wasteform from freeze thaw deterioration. Most other degradation mechanisms are initiated by the flux of species to the vaults, such that the degradation would occur at exposed surfaces. Because of the low hydraulic and effective diffusivity properties of the initial materials, it is unlikely that mechanisms that result in a shrinking core type of attack would result in an increase of the bulk hydraulic conductivity of the wasteform to greater than 1×10^{-7} cm/s in 10,000 years. However, this conclusion is based on short-term observations and tests of materials that may not be sufficiently analogous to the saltstone wasteform. Additional studies by DOE to justify the assumed long-term degradation rates of the wasteform may significantly reduce uncertainty.

In this TER, the NRC staff has identified factors important to assessing compliance with 10 CFR 61, Subpart C, including improvements in future modeling and the associated support to justify the predictions of the modeling.

<u>NRC staff used the following assumptions in assessing conformance with Criterion Three (A)</u>:

1. More realistic modeling of waste oxidation and release of technetium from an oxidized layer of waste will result in predicted doses significantly lower than those projected in the DOE sensitivity analysis for 100% oxidation of the waste (see Section 4.2.9).

2. The hydraulic conductivity of degraded saltstone and vault concrete will not be larger than 1×10^{-7} cm/s (1×10^{-1} ft/yr) (see Section 4.2.7).

3. Field-scale physical properties (e.g., hydraulic conductivity, effective diffusivity) of as-emplaced saltstone are not significantly different from the results of laboratory tests of smaller-scale samples performed to date (see Section 4.2.7).

4. Cracking from any mechanism will not be significantly more extensive than currently predicted by DOE (see Sections 4.2.6 and 4.2.9).

5. The overall numerical modeling results for moisture flow through fractures in the concrete and saltstone located in the vadose zone will be confirmed by future model support (i.e., fracture flow will not occur) (see Section 4.2.9).

6. The previous DSA bounds the impacts to individuals during operations that may be estimated from the proposed waste streams (see Section 4.2.20).

7. Active institutional controls will be maintained for 100 years (see Section 4.2.18).

8. Current projections of the radiological concentrations of the waste are greater than or equal to actual concentrations for highly radioactive radionuclides (see Section 4.2.3).

9. Future tests will confirm that the physical properties of samples that contain organic materials similar to Tank 48 waste are consistent with non-organic containing samples (see Section 4.2.7).

10. The erosion control design that DOE eventually implements will not deviate significantly from the information submitted to the NRC in (WSRC, 2005a) and the associated references (see Section 4.2.5.1).

11. Development and use of accurate information for the moisture characteristic curves of concrete and saltstone will not significantly increase currently estimated release rates (see Section 4.2.9).

12. Gas phase transport of oxygen will not significantly increase oxidation of technetium in the saltstone (including transport in fractures) (see Section 4.2.9).

<u>NRC staff concludes the following with respect to Criterion Three (A)</u>:

1. There is reasonable assurance that the performance objective of 10 CFR 61.41 can be met, including the provision of ALARA releases of radioactivity to the general environment assuming that the aforementioned assumptions relevant to the performance objective are verified.

2. There is reasonable assurance that the performance objective of 10 CFR 61.42 for protection of individuals from inadvertent intrusion can be satisfied, given that a design for long-term erosion control is implemented which greatly reduces the likelihood of an agricultural intruder scenario occurring.

3. There is reasonable assurance that the performance objective of 10 CFR 61.43 for protection of individuals during operations can be met. During operations, individuals are protected by DOE regulations which were demonstrated to provide protection

comparable to 10 CFR Part 20. In addition, a number of measures are applied to ensure that exposure of individuals are maintained as low as reasonably achievable including: (1) a documented radiation protection program, (2) a Documented Safety Analysis, (3) design of the SPF and SDF, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, and dosimetry.

4. DOE's plans to place the saltstone grout wasteform in concrete vaults eventually covered by a thick, engineered cap specifically designed for erosion control are sufficient to indicate that there is reasonable assurance that the performance objective of 10 CFR 61.44 (Stability of the Disposal Site After Closure) can be met.

4.3.1 Factors Important to Assessing Compliance with 10 CFR 61, Subpart C

In general, verification of the assumptions made in assessing whether the performance objectives of 10 CFR 61, Subpart C, can be met should be performed by DOE. However, some of the assumptions made in the analysis, if incorrect, could lead to noncompliance with the performance objectives. It is these types of assumptions that the NRC plans to monitor as part of its responsibilities under the NDAA. These assumptions fall into the following general groups: wasteform and vault degradation, the effectiveness of infiltration and erosion controls, and estimation of the radiological inventory. The NRC staff concludes that for this draft waste determination, the following factors are important to assessing whether DOE's disposal actions will be compliant with the performance objectives of 10 CFR 61, Subpart C:

1. The rate of waste oxidation and release of technetium from an oxidized layer of saltstone will be a key determinant of the future performance of the saltstone disposal facility and therefore whether 10 CFR 61.41 can be met. More realistic modeling will be important to achieving the performance objectives, and adequate model support is essential to providing the technical basis for the model results. It will be important to ensure that gas phase transport of oxygen through fractures will not significantly increase oxidation of technetium in the saltstone.

2. The extent of degradation that may influence the hydraulic isolation capabilities of the saltstone and vaults will be a key factor in assessing whether the SDF can meet 10 CFR 61.41. Degradation mechanisms that may result in the hydraulic conductivity of degraded saltstone and vault concrete being larger than 1×10^{-7} cm/s (1×10^{-1} ft/yr) need to be evaluated with multiple sources of information (e.g., modeling, analogs, experiments [especially field scale and long-term], expert elicitation) to ensure that they are unlikely to occur. It will be important to ensure that field-scale physical properties (e.g., hydraulic conductivity, effective diffusivity) of as-emplaced saltstone are not significantly different from the results of laboratory tests of smaller-scale samples performed to date. It will be important to perform additional laboratory measurements of hydraulic conductivity because the data being relied upon represent limited samples that had a small range of curing times. In addition, because there was a fairly significant amount of variability in the TCLP test results, if DOE deviates significantly from the nominal saltstone composition, DOE should perform additional tests for hydraulic conductivity and effective diffusivity that justify the parameter values used over the range of compositions.

3. Adequate model support is essential to assessing whether the saltstone disposal facility can meet 10 CFR 61.41. The model support for: (1) moisture flow through fractures in the concrete and saltstone located in the vadose zone, (2) realistic modeling of waste oxidation and release of technetium, (3) the extent and frequency of fractures in saltstone and vaults that will form over time, (4) the plugging rate of the lower drainage layer of the engineered cap, and (5) the long-term performance of the engineering cap as an infiltration barrier is key to confirming performance assessment results.

4. The erosion control design is important to ensuring that 10 CFR 61.42 can be met because it eliminates pathways and scenarios for intruder dose assessments. Implementation of an adequate design that does not deviate significantly from the information submitted to the NRC in (WSRC, 2005a) and the associated references is important, or if it does deviate significantly that it is reviewed by NRC staff to ensure that the revisions are consistent with long-term erosion control design principles.

5. The infiltration control design is important to ensuring that 10 CFR 61.41 can be met because the release of contaminants to the groundwater is predicted to be sensitive to the large reduction in infiltration provided by the infiltration control. It is important to ensure that the design can be implemented and will perform as designed.

6. Implementation of an adequate sampling plan is important to ensuring that 10 CFR 61.41 and 10 CFR 61.42 can be met. It is important to assess results of future sampling and confirm that current projections of the concentrations of highly radioactive radionuclides in treated salt waste (or grout) are greater than or equal to actual concentrations of highly radioactive radionuclides in treated salt waste (or grout).

7. To ensure that Tank 48 waste can be safely managed, future tests of the physical properties of samples that contain organic materials similar to Tank 48 waste will need to confirm that the properties of the wasteform made from this waste will provide for suitable wasteform performance such that the disposal system will be able to meet the performance objectives. The technical basis should, at a minimum, include tests for hydraulic conductivity and effective diffusivity.

8. Predicted removal efficiencies of highly radioactive radionuclides by each of the planned salt waste treatment processes are a key factor in determining the radiological inventory disposed of in saltstone. The inventory, in turn, is a important factor in the determination that 10 CFR 61.41 and 10 CFR 61.42 can be met.

This is not an all-inclusive list of factors that may need to be monitored by the NRC to assess compliance with the performance objectives, but rather is based on DOE's current planned approach and NRC's current analysis of DOE's approach. Therefore, the factors that need to be monitored may change as DOE implements its disposal plans. A monitoring plan of DOE's disposal actions for the waste assessed in this draft waste determination will be developed, in coordination with the State, that will present the details of NRC's planned future monitoring activities.

5. CONCLUSIONS

As discussed in detail in previous sections, the NRC staff has conducted a technical analysis of DOE's draft waste determination for salt waste disposal at the SRS. NRC staff concludes that there is reasonable assurance that DOE can meet the criteria provided in the NDAA in Section 3116 (a)(1), (a)(2), and (a)(3)(A)(i). This conclusion is based on information presented in DOE's draft Section 3116 waste determination dated March 2005, DOE's responses to NRC's RAI, supporting references, and information provided during meetings between NRC and DOE. If, in the future, DOE determines it is necessary to revise its assumptions, analysis, design, or waste management approach and those changes are important to meeting the criteria of the NDAA, DOE should consult once again with the NRC regarding the conclusions of this TER. It should be noted that NRC staff is providing consultation to DOE as required by the NDAA and the NRC is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW. This NRC assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental waste determinations at SRS or at other sites.

DOE should assess the factors that the NRC staff have identified as important to assessing compliance with 10 CFR 61, Subpart C, and carefully review the assumptions that the NRC conclusions are based upon. The DOE analysis demonstrates that more realistic modeling of infiltration, water contact with the waste, waste oxidation, and radionuclide release in an unsaturated and potentially fractured system is needed, because conservative modeling does not yield acceptable results. Likewise, adequate model support is essential to confirm the model results and to ensure that public health and safety can be protected with reasonable assurance. The deterministic DOE base case result does not have adequate model support and is not conservative. Considering the uncertainty in many key parameters, it should not be used as the basis for developing inventory limits. A revised base case should be based on the projected average vault inventory and orientation of multiple disposal vaults, include the expected magnitude and timing of climate change from the natural cycling of climates, include the expected magnitude and rate of oxidation of waste, consider liquid and gas flow in fractures that may develop, and account for the questionable moisture characteristic curve information for concrete and saltstone that was used in the previous analysis.

APPENDIX A.   NRC STAFF RECOMMENDATIONS

During its consultative review of the "Draft Section 3116 Determination, Salt Waste Disposal, Savannah River Site" dated March 2005, the NRC staff made some observations regarding DOE's waste management approach. As a result of these observations, the NRC staff has developed some recommendations for DOE's consideration. The purpose of the recommendations is to communicate actions that DOE might consider to further enhance its approach for management of the salt waste at SRS, as well as the approach for future waste determinations. As stated in this TER, the NRC staff has concluded that there is reasonable assurance that DOE will meet the applicable criteria of the NDAA if certain assumptions made in DOE's analyses are verified. Thus, it is the NRC staff's view that implementation of these recommendations is not necessary to meet the criteria in the NDAA. The recommendations are based on the information provided in the TER and a more detailed discussion of the underlying bases for the recommendations can be found in the referenced sections.

1. DOE should continue to base identification of highly radioactive radionuclides in part on the results of sensitivity or uncertainty analyses (see Section 3.4).

2. If warranted by operating experience, DOE should consider operational practices or process modifications to minimize sludge carryover and entrainment in salt and supernate during the DDA process (see Section 3.6).

3. Because DOE predicts salt treatment will continue at SRS until at least 2019 and because DOE estimates that only approximately 1% of the Se-79, Tc-99, and I-129 that will be disposed of in the SDF will result from DDA waste (d'Entremont and Drumm, 2005), DOE may have the opportunity to implement processes to improve removal of these radionuclides. Processes that remove these radionuclides, which are expected to dominate doses to members of the public from the SDF, could reduce long-term risk to the public even if they are implemented several years after the proposed approach has begun (see Sections 4.2.15 and 4.2.16). Therefore, DOE should consider further evaluation of the practicality of additional removal of Se-79, Tc-99, and I-129 from SRS salt waste (e.g., performing additional basic research to develop technologies for the removal of Se-79, Tc-99, and I-129) to limit the inventory of long-lived radionuclides and the uncertainties in dose resulting from uncertainty in the long-term performance of the disposal system (see Section 3.8).

4. DOE should continue to assess the balance between removal of radionuclides that cause worker risk and are short-lived (e.g., Cs-137) and removal of radionuclides that cause potential future public risks and are long-lived (e.g., Se-79, Tc-99, I-129) in future radionuclide removal strategies (see Sections 3.5, 3.8, and 4.2.16).

5. DOE should consider adopting a probabilistic approach for future performance assessments (see Sections 4.2 and 4.2.16).

6. In developing its sampling plan, DOE may want to consider how adjusting the inventory limits could assist in establishing the appropriate number of samples needed to confirm the actual waste sent to the SPF does not exceed the WAC (see Section 4.2.3).

7. DOE should consider addressing the degradation of the lower drainage layer through further studies to reduce uncertainty or through additional conservatism in cap design (see Section 4.2.5).

8. To help reduce uncertainties, DOE should consider performing additional laboratory measurements of initial hydraulic conductivity, as well as long-term tests or monitoring studies designed to evaluate the long-term durability of the saltstone and concrete vault (see Section 4.2.7).

9. Unless mitigated by future design modifications, DOE should explicitly consider rebar and fill pipe corrosion on the integrity and projected lifetime of the vaults in future degradation calculations. In the response to action items, DOE indicated that if penetrations in the vault provide an unacceptable moisture flow pathway, the closure plan for the facility will be revised to provide modifications to the design (see Section 4.2.7).

10. In developing model support to justify the assumption of lack of flow through fractures, DOE should consider: (1) heterogeneity in material properties, (2) temporal variations in saturation, especially resulting from rapid transport of infiltration through root holes to the lower drainage layer, (3) infilling of fractures and joints with porous media, (4) offset of the essentially impermeable vault/saltstone on either sides of the fractures, (5) variability in the aperture of the joints and fractures, (6) sensitivity to grid size in the simulations, especially at the interface of the fracture and porous media, and (7) variability in moisture characteristic curve parameters (see Sections 4.2.9 and 4.3.1).

11. In future performance assessments, DOE should consider the impact of the use of steady-state flow fields and the impact of fluid densities on estimated release rates (see Section 4.2.9).

12. In saturated zone modeling, DOE should consider reducing the grid spacing in the portion of the model immediately surrounding Vault 4 and the compliance nodes to verify that peak concentrations are not being arbitrarily diluted by the grid discretization (see Section 4.2.11).

13. DOE should consider that averaging contaminant releases over the footprint of the vault may not be realistic for scenarios with discrete releases (e.g., flow in widely-spaced fractures) (see Section 4.2.11).

14. If the performance assessment is significantly changed in the future, DOE should develop additional technical basis for the uranium K_d to support the selected value (see Section 4.2.11).

15. If subsequent calculations suggest that dilution at the water table may be less than previously predicted, DOE should consider potential chemical effects on saturated zone sorption coefficients (see Section 4.2.11).

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LIST OF ABBREVIATIONS AND ACRONYMS

AFP	Alpha Finishing Process
ALARA	As Low As Is Reasonably Achievable
Am	americium
ARP	Actinide Removal Process
ASP	Alpha Strike Process
Bq	becquerel
Bq/mL	becquerel/milliliter
C	carbon
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
Cs	cesium
CSSX	Caustic Side Solvent Extraction
DDA	Deliquification, Dissolution, and Adjustment
DNFSB	Defense Nuclear Facilities Safety Board
DOE	U.S. Department of Energy
DOE-SRS	U.S. Department of Energy, Savannah River Site
DUST-MS	Disposal Unit Source Term – Multiple Species
DWPF	Defense Waste Processing Facility
EPA	U.S. Environmental Protection Agency
HLW	high-level radioactive waste
I	iodine
IA	Interagency Agreement
ICRP	International Commission on Radiological Protection
in.	inches
ITP	In-Tank Precipitation
GCL	geosynthetic clay layer
K_d	distribution coefficient
K_h	hydraulic conductivity
L	liters
LLW	low-level radioactive waste

NDAA	National Defense Authorization Act of Fiscal Year 2005
m	meters
m^3	cubic meters
MCi	million curies
MCU	Modular CSSX Unit
mL/g	milliliters/gram
MOU	Memorandum of Understanding
mrem	millirem
MST	Monosodium Titanate
mSv	millisievert
NAS	National Academy of Sciences
Np	neptunium
NRC	U.S. Nuclear Regulatory Commission
PA	performance assessment
pCi/mL	picocuries/milliliter
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
Pu	plutonium
RAI	request for additional information
SDF	Saltstone Disposal Facility
SPF	Saltstone Production Facility
Sr	strontium
SRM	Staff Requirements Memorandum
SRS	Savannah River Site
Sv	sievert
SWPF	Salt Waste Processing Facility
TBq	terabecquerel
TPB	tetraphenylborate
Tc	technetium
TCLP	Toxicity Characteristic Leaching Procedure
TEDE	Total Effective Dose Equivalent
TER	Technical Evaluation Report
TRU	transuranic

WAC	Waste Acceptance Criteria
WCS	Waste Characterization System
WIR	waste-incidental-to-reprocessing
yr	year