U.S. NUCLEAR REGULATORY COMMISSION PLAN FOR MONITORING THE U.S. DEPARTMENT OF ENERGY SALT WASTE DISPOSAL AT THE SAVANNAH RIVER SITE IN ACCORDANCE WITH THE NATIONAL DEFENSE AUTHORIZATION ACT FOR FISCAL YEAR 2005

May 3, 2007

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ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ASTM	American Society of Testing and Materials
ARP	Actinide Removal Process
DDA	Deliquification, Dissolution, and Adjustment
DOE	U.S. Department of Energy
EPA	U.S. Environmental Protection Agency
ESOP	Environmental Surveillance and Oversight Program
GCL	Geosynthetic Clay Liner
HELP	Hydrologic Evaluation of Landfill Performance
HLW	High-Level Waste
HRR	Highly Radioactive Radionuclide
MCU	Modular Caustic Side Solvent Extraction Unit
NDAA	National Defense Authorization Act
NRC	U.S. Nuclear Regulatory Commission
PA	Performance Assessment
PMP	Probable Maximum Precipitation
PODD	DOE's Performance Objective Demonstration Document
SCDHEC	South Carolina Department of Health and Environmental Control
SDF	Saltstone Disposal Facility
SRP	Standard Review Plan for Activities Related to U.S. Department of Energy Waste Determinations (NUREG–1854)
SPF	Saltstone Production Facility
SRS	Savannah River Site
SWPF	Salt Waste Processing Facility
TCLP	Toxicity Characteristic Leaching Procedure
TER	Technical Evaluation Report
TPB	Tetraphenylborate
WSRC	Westinghouse Savannah River Company

DEFINITIONS

Closed activity	A monitoring activity where a key assumption made or key parameter used by the U.S. Department of Energy (DOE) in its assessment has been either substantiated or determined not to be important in meeting the performance objectives of 10 CFR Part 61, Subpart C.
Disposal	The isolation of radioactive wastes from the biosphere.
Factors	Assumptions made or parameters used by DOE in its performance demonstration that U.S. Nuclear Regulatory Commission (NRC) considers important, through the review of DOE waste determination which describes its waste disposal actions, and demonstrates that performance objectives listed in 10 CFR Part 61, Subpart C, will be met.
Highly radioactive radionuclides	Those radionuclides that contribute most significantly to risk to the public, workers, and the environment.
Monitoring activities	NRC and state activities to assess compliance with the performance objectives listed in 10 CFR Part 61, Subpart C.
Noncompliance	A conclusion that DOE’s disposal actions does not result in compliance with the performance objectives of 10 CFR Part 61, Subpart C, or that there is an insufficient basis to assess whether DOE’s disposal actions will result in compliance.
Onsite	Areas of the DOE site where monitoring activities will be carried out. This may include areas of the site outside of the immediate Saltstone Production Facility and Saltstone Disposal Facility area.
Open activity	Monitoring activity that has not been closed and for which sufficient information has not been obtained to fully assess compliance with a 10 CFR Part 61, Subpart C, performance objective.
Open-noncompliant activity	An ongoing monitoring activity that has provided evidence that the performance objectives of 10 CFR Part 61, Subpart C, are currently not being met or will not be met in the future or for which insufficient competent technical bases have been provided to determine that the performance objectives will be met.

DEFINITIONS (continued)

Operations	The timeframe during which DOE carries out its waste disposal actions, through the end of the institutional control period. For the purpose of this plan, DOE actions involving waste disposal are considered to include performance assessment development (analytical modeling), waste removal, grouting, stabilization, observation, maintenance, or other similar activities.
Performance assessment	A type of systematic (risk) analysis that addresses (a) what can happen, (b) how likely it is to happen, (c) what the resulting impacts are, and (d) how these impacts compare to regulatory standards.
Performance objectives	NRC 10 CFR Part 61, Subpart C, requirements for low-level waste disposal facilities that include protection of the general population from releases of radioactivity (10 CFR 61.41), protection of individuals from inadvertent intrusion (10 CFR 61.42), protection of individuals during operations (10 CFR 61.43), and stability of the disposal site after closure (10 CFR 61.44).
Substantiate	To establish by competent technical bases.
Waste determination (or non-high-level waste determination)	DOE documentation to demonstrate that a specific waste stream is not high-level waste.
Worker	DOE personnel or contractors who carry out operational activities at the disposal facility. For the purpose of this plan, 10 CFR Part 835 dose limits (comparable to 10 CFR Part 20) would apply for radiation workers.

EXECUTIVE SUMMARY

The National Defense Authorization Act for Fiscal Year 2005 (NDAA) gives the U.S. Department of Energy (DOE) the authority to determine whether certain waste resulting from the reprocessing of spent nuclear fuel is high-level waste (HLW) that requires geologic disposal or non-HLW suitable for near-surface disposal. Section 3116 of the NDAA requires DOE to consult with the U.S. Nuclear Regulatory Commission (NRC) regarding DOE's non-HLW determinations performed pursuant to the NDAA. In addition, Section 3116 of the NDAA requires NRC to monitor DOE's disposal actions to assess compliance with the performance objectives established in NRC's licensing requirements for land disposal of radioactive waste (10 CFR Part 61, Subpart C). These performance objectives include provisions for the protection of the general population from releases of radioactivity, the protection of individuals from inadvertent intrusion, the protection of individuals during operations, and the stability of the disposal site after closure.

In March 2005, DOE provided NRC with a draft waste determination that described DOE's plan to dispose of salt waste at the Savannah River Site (SRS) (DOE, 2005a). In its draft waste determination, DOE proposed to remove salt waste from the 49 operational waste tanks at SRS, treat the waste with various processes to remove some of the radionuclides in the waste, and solidify the treated salt waste by mixing it with dry grout ingredients to form a cementitious wasteform called "saltstone." Based on a review of DOE's draft waste determination and supporting documents, NRC staff concluded that there is reasonable assurance that the salt waste will meet the applicable NDAA criteria if several key assumptions are verified during monitoring. This conclusion and NRC staff's review are documented in NRC's technical evaluation report (TER) for salt waste disposal at SRS (NRC, 2005a). This document describes NRC's plan to monitor the disposal of salt waste at SRS to fulfill its responsibilities under Section 3116 of the NDAA.

One part of NRC's monitoring plan is to monitor radionuclide concentrations in the air and groundwater at SRS. This type of monitoring often is referred to as "environmental monitoring." Because DOE has a history of performing environmental monitoring at SRS, NRC staff plans to review environmental monitoring data DOE provides in its reports to fulfil its own monitoring program requirements (see Sections 3.1.1.3 and 5.1.2). To develop confidence that DOE's environmental monitoring data are appropriate for use in NRC's monitoring of salt waste disposal at SRS, NRC staff plans to review DOE's environmental monitoring procedures and to observe DOE's groundwater and air sampling activities during onsite observations (see Sections 3.2.6 and 5.2.2).

In addition to environmental monitoring, NRC staff will monitor specific technical areas that it identified as important to assessing compliance with the performance objectives during its review of DOE's draft waste determination. NRC's TER describes key assumptions DOE made in its analyses supporting its salt waste determination and the resulting technical areas, called "factors" in this report, that NRC staff plans to monitor to assess compliance with the performance objectives. NRC staff identified the following eight key factors to monitor: (i) oxidation of saltstone, (ii) hydraulic isolation of saltstone, (iii) model support, (iv) erosion control design, (v) infiltration barrier performance, (vi) feed tank sampling, (vii) Tank 48 wasteform, and (viii) radionuclide removal efficiencies. Each of these factors is summarized in

the Appendix of this plan. In general, the factors relate to three important aspects of the disposal system: wasteform and vault degradation, the effectiveness of infiltration and erosion controls, and estimation of the radiological inventory.

As explained in the TER, NRC based its assessment of compliance on a 10,000-year performance period. Because of the long performance period, several of the monitoring factors relate to the long-term degradation of saltstone and the concrete vaults that the saltstone will be poured into. Chemical oxidation of saltstone was identified as a monitoring factor primarily because of the possibility of unacceptable technetium doses if saltstone is oxidized more rapidly than DOE predicts. To confirm DOE's assumptions about saltstone oxidation, NRC staff expect to monitor the development of better predictions of saltstone oxidation during the 10,000-year performance period and the resulting release of technetium. Specifically, staff expects to monitor the results of oxidation experiments and refined radionuclide release models, among other possible activities (see Section 3.1.2).

Physical degradation of saltstone is expected to affect facility performance because more water can flow through a degraded wasteform than an intact wasteform, and increased water flow through the wasteform is expected to increase radionuclide releases to groundwater. Thus the physical degradation of saltstone during the 10,000-year performance period is of interest primarily because degradation is expected to compromise the hydraulic isolation of the waste. Two important aspects of NRC's plan to monitor the hydraulic isolation of saltstone are (i) to confirm that the hydraulic properties of saltstone at the disposal site are consistent with the properties of the laboratory samples of saltstone described in the waste determination and (ii) to monitor the development of better predictions of saltstone degradation over long time periods (see Section 3.1.3). Waste in one of the tanks, Tank 48, is unlike the rest of the salt waste at SRS because it contains a substantial amount of organic salts; as a result, NRC staff expects to monitor the hydraulic properties and long–term degradation of saltstone made from this waste as a separate monitoring factor (see Section 3.1.7).

Infiltration and erosion controls are both part of an engineered cap that DOE plans to use to cover the saltstone disposal facility at facility closure. The infiltration control system was identified as a factor for monitoring because the predicted dose to a potential member of the public was sensitive to DOE's assumption that the infiltration control system would significantly limit the amount of water reaching the waste for the entire 10,000–year performance period. To monitor the design and performance of the infiltration control system, NRC staff expect to verify that the infiltration controls are implemented as described in the waste determination and supporting documents or that any changes made to the design do not degrade facility performance. Specifically, if the design is not changed, NRC staff expects to monitor the development of information to support assumptions DOE made about the rate at which the lower drainage layer in the infiltration system would become plugged and any information developed to support the performance of the cap as an infiltration barrier (See Section 3.1.5).

Implementation of an adequate erosion control design is important to protecting a potential inadvertent intruder, because the erosion control barrier will help to maintain a thick layer of soil over the vaults, which reduces the potential for intrusion into the waste. The primary activity the staff plan to perform to monitor the implementation of the erosion control design is to verify that the erosion control barrier is built as DOE described to NRC during consultation or that, if changes are made to the design, the new design will be as effective in limiting erosion as the design described in documents used to support the waste determination (See Section 4.1.2).

Two monitoring factors relate to the final inventory of radionuclides in the saltstone disposal facility. In general, total inventories of radionuclides in the salt waste disposal facility are expected to affect the potential dose to a hypothetical offsite receptor, while concentrations of radionuclides in the waste are expected to significantly affect the potential dose to a hypothetical intruder who comes into direct contact with the waste. Because of the importance of the saltstone inventory and wasteform concentrations, staff plan to monitor reported disposal site inventories as well as sampling of the salt waste preparation feed tank to assess whether the inventory and concentrations of radionuclides sent to the saltstone disposal facility are consistent with the inventories and concentrations that DOE used as a basis for their waste determination (see Sections 3.1.1.1 and 3.1.6). Similarly, the staff expect to monitor how well each of the planned salt waste treatment processes remove radionuclides from the waste, because removal of radionuclides from the waste will affect the inventory of radionuclides in the salt waste disposal facility. In addition, staff will monitor radionuclide removal to assess whether potential doses to members of the general public will be maintained as low as is reasonably achievable (ALARA), as required by the performance objective for protection of the general public from releases of radioactivity (see Sections 3.1.1.2 and 3.1.8).

In addition to these specific factors, the NRC staff also plans to monitor the development of model support in several technical areas. Essentially, model support provides assurance that the results of any models used to predict potential doses or intermediate results of submodels are consistent with independent data. In the TER, NRC staff indicated it would monitor the development of model support in the following technical areas: (i) moisture flow through fractures in the concrete and saltstone located in the vadose zone, (ii) realistic modeling of waste oxidation and release of technetium, (iii) the extent and frequency of fractures in saltstone and vaults that will form over time, (iv) the plugging rate of the lower drainage layer of the engineered cap, and (v) the long-term performance of the engineering cap as an infiltration barrier. Each of these areas is related to other monitoring factors. However, the "model support" monitoring factor is different from the other factors because its goal is to provide confidence in aspects of the model or models used to make dose predictions. Thus to monitor model support development, NRC staff expects to compare available data about the development of the disposal system or analogous systems with model predictions. Ideally, model support includes multiple lines of evidence supporting the conclusions of modeled dose predictions or intermediate submodels, such as radionuclide release or transport in the subsurface. Lines of evidence may include site characterization and design data, results of process-level modeling, laboratory testing, field measurements, analogs, and formal independent peer review (see Section 3.1.4).

This monitoring plan describes how NRC staff plans to assess compliance with each of the performance objectives and specifically how staff plans to evaluate each of the monitoring factors through technical analyses and onsite observations. In addition, this plan describes the actions NRC staff plans to take if it concludes that the disposal actions are not in compliance with the performance objectives (see Section 7.3). The NDAA specifically cites the performance objectives in NRC's licensing requirements for land disposal of radioactive waste (10 CFR Part 61, Subpart C). Because the performance objectives include a general requirement that there is reasonable assurance that exposures to humans will not exceed the limits established in 10 CFR 61 Subpart C, DOE's disposal actions would be out of compliance if NRC no longer had reasonable assurance that the performance objectives would be met. If NRC concludes that DOE waste disposal actions are not in compliance with the performance objectives, NRC will, in accordance with NDAA, Section 3116 requirements, notify Congress,

the State, and DOE. Because NRC does not have regulatory or enforcement authority over DOE, it is the role of Congress, the State, and DOE to determine what, if any, actions will be taken in response to a noncompliance report.

1 INTRODUCTION

1.1 Background

To initiate consultation activities for salt waste disposal at the Savannah River Site (SRS) in South Carolina, on March 31, 2005, United States Department of Energy (DOE) submitted its draft waste determination under the National Defense Authorization Act for Fiscal Year 2005 (NDAA) (DOE, 2005a) to United States Nuclear Regulatory Commission (NRC). NRC concluded that there is reasonable assurance that the applicable criteria of the NDAA could be met for the salt waste disposal in its technical evaluation report (TER) dated December 28, 2005 (NRC, 2005a). In the TER, NRC staff identified several key assumptions DOE made in its performance assessment (PA) that are important to demonstrating compliance with performance objectives listed in 10 CFR Part 61, Subpart C. These key assumptions, which are called “factors,” are listed and described in Section 3.3.1 of the staff TER (see Appendix A). The staff considered substantiation of these key assumptions to be important to DOE’s compliance demonstration and consequently, to NRC’s monitoring responsibilities under the NDAA.

On January 17, 2006, DOE issued its final waste determination for salt waste disposal at the SRS (DOE, 2006).[1] The disposal plans described in the final waste determination were, generally consistent with plans described in the draft waste determination. The specific technical guidance in this monitoring plan is based on the analyses of DOE’s draft waste determination as documented in NRC’s TER. Consequently, changes that DOE may have made to the disposal plan since the March 31, 2005, submittal of the draft waste determination are not necessarily reflected in this document. NRC staff expects to update this monitoring plan as necessary to reflect changes in DOE’s disposal plans.

1.2 Objective

This document describes the activities that NRC will perform to monitor DOE actions related to disposal of salt waste at SRS. The focus of the monitoring may need to be unique and specifically tailored for each waste determination. This document includes NRC plans for monitoring salt waste disposal at SRS; additional general monitoring information can be found in NRC’s guidance document for consultation activities.

1.3 Interfacing With the State of South Carolina

For DOE’s waste determination for the SRS salt waste disposal, the staff began interacting with the State of South Carolina during the consultation phase with DOE. Through discussions with staff at the South Carolina Department of Health and Environmental Control (SCDHEC), it was established that the State has two primary regulatory responsibilities for the salt waste disposal: (i) a wastewater permit for the salt waste processing facility (SPF) and (ii) an industrial solid waste permit for the salt waste disposal facility (SDF).

[1]For the first two non-HLW determinations under the NDAA, DOE provided NRC staff a draft waste determination for NRC technical review as part of the consultation process. Following completion of NRC staff’s technical evaluation, DOE finalized the waste determinations considering NRC’s analyses and technical comments documented in the TERs.

The wastewater permit, which is issued by the Bureau of Water within SCDHEC, establishes approximate limits on the chemical composition of the salt waste feed. It requires quarterly sampling of the salt waste feed for chemical constituents and semiannual sampling for radionuclides. It also requires DOE to record on a daily basis the amount of salt waste processed and the amount of saltstone produced. DOE is currently limited to sending no more than 530 m^3 per day [140,000 gal per day] of saltstone solution to the SDF.

The solid waste permit, which is issued by the Bureau of Land and Waste Management within SCDHEC, requires the installation and operation of a groundwater detection monitoring system for the SDF. The original groundwater monitoring system consisted of one upgradient and one downgradient well location for Vault 1. The current plan calls for maintaining the upgradient well location, but also for the installing of at least three downgradient wells from each vault. DOE is required to sample for a suite of constituents on a semi-annual basis, including gross alpha, gross beta, gamma emitters, I-129, and H-3. DOE also must sample for Ra-226, Ra-228, and Tc-99 on a biennial basis. Samples collected semiannually are to be reported on an annual basis, while biennial samples are reported every two years. DOE is also required to include information on the groundwater flow direction and flow rate within its annual report.

In discussions with SCDHEC Solid Waste Hydrogeology Section, the State indicated that it does not collect split samples from DOE wells; however, the State does occasionally observe DOE sample collection. Even though the State does not routinely observe the installation of wells, the wells are required to meet specific construction requirements as specified in the State regulations. Further, the wells must be installed by a State-certified well driller, and the driller is required to submit a well completion report to the SCDHEC Bureau of Water. To fulfill its groundwater monitoring responsibilities, DOE has developed a groundwater monitoring plan for the SDF (DOE, 2005b).

The SCDHEC solid waste permit does not require any additional monitoring other than groundwater; however, SCDHEC staff indicated that DOE must submit as-built drawings of the vaults' construction to confirm that they have been built as designed. The as-built drawings must be stamped by a state-licensed professional engineer. The permit also requires at least 30 years of postclosure facility care. Thus, DOE is required to maintain surveillance of the facility for at least 30 years after closure.

In addition to the environmental monitoring that DOE must undertake, the State's environmental surveillance and oversight program (ESOP) is an extensive monitoring program that covers the whole SRS and beyond. The ESOP was established by SCDHEC in 1995 as part of its Agreement In Principal program with DOE, focusing on providing an independent, but nonregulatory, evaluation of the site. Its main focus is to provide the public with an independent source of information on the effectiveness of DOE's monitoring activities. Through a network of monitoring sites, radiological constituents are analyzed from samples collected in air, groundwater, surface water, drinking water, fish, game animals, aquatic insects, edible and nonedible vegetation, soil, and milk. Results are published in an annual report. Currently, the annual report is available upon request; however, SCDHEC hopes to eventually make the reports available for internet download. Although the ESOP monitoring results are not intended to be used for the State's regulatory purposes, the results can be used to gauge the effectiveness of DOE's environmental monitoring.

2 MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.40—GENERAL REQUIREMENT

Section 3116 (a)(3)(A) and (B) of the National Defense Authorization Act states that for radioactive waste from reprocessing of spent nuclear fuel to be determined to be non-high-level waste, it must be shown that it will be disposed of in compliance with the performance objectives in Subpart C of Part 61 of Title 10, Code of Federal Regulations. Subpart C of 10 CFR Part 61 requires that disposal facilities must be sited, designed, operated, closed, and controlled after closure to ensure compliance with the performance objectives in 10 CFR 61.41 through 61.44. To assess compliance with the requirements of 10 CFR 61.40 for the Savannah River Site salt waste disposal, the U.S. Nuclear Regulatory Commission will rely upon its assessment of the U.S. Department of Energy (DOE) compliance with 10 CFR 61.41 through 61.44. Specifically, DOE will be viewed as in compliance with 10 CFR 61.40 as long as it is viewed as in compliance with the other performance objectives.

3 MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.41—PROTECTION OF GENERAL POPULATION

10 CFR 61.4. Protection of the general population from releases of radioactivity.

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

The primary focus of the U.S. Nuclear Regulatory Commission (NRC) monitoring activities to assess compliance with 10 CFR 61.41, especially while waste disposal activities are ongoing, will be to ensure that the U.S. Department of Energy (DOE) performance assessment (PA) demonstrates that the performance objective can be met. Other aspects of monitoring are intended to provide an early indication of the facility performance and to ensure compliance with the as low as reasonably achievable (ALARA) provision and long-term monitoring. NRC plans to carry out three primary activities to assess compliance with 10 CFR 61.41:technical reviews (Section 3.1), onsite observations (Section 3.2), and long-term monitoring (Section 3.3).

3.1 Technical Reviews

NRC plans to carry out three primary technical review activities to assess compliance with 10 CFR 61.41: reviewing data (Section 3.1.1), reviewing studies and analyses to address key factors identified in the technical evaluation report (TER) (Sections 3.1.2–3.1.8), and reviewing changes to DOE PA (Section 3.1.9).

3.1.1 Data Reviews

3.1.1.1 Radioactive Inventory

To demonstrate that the Saltstone Disposal Facility (SDF) will meet the 10 CFR 61.41 performance objective, DOE compared the entire projected SDF inventory to inventory limits for a single vault using a sum-of-fractions approach and showed that the sum-of-fractions was less than one (Rosenberger, et al., 2005).[2] In the TER, NRC staff concluded that, if an adequate sampling plan was implemented, it would be appropriate for DOE to use a sum-of-fractions approach with the actual saltstone inventory to demonstrate compliance with the performance objectives. The sum-of-fractions approach is based on the assumption that there is a direct relationship between a radionuclide's inventory and the dose the radionuclide could cause to a hypothetical receptor. Thus, radionuclide inventory is expected to directly affect DOE's demonstration that the performance objective of 10 CFR 61.41 can be met.

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[2] The inventory limits DOE used to make this comparison were limits for SDF vault 4 that DOE developed in a special analysis (Cook, et al., 2005). DOE did not present separate inventory limits for the entire SDF.

DOE predicts that Cs-137 and its short-lived daughter Ba-137m will represent 96 percent of the projected radioactivity in the SDF. While Cs-137/Ba137m are expected to dominate the dose to a worker or to a hypothetical receptor who intrudes into the waste at approximately 100 years after closure, other radionuclides are expected to dominate the dose to a hypothetical receptor who lives near the disposal site but does not intrude into the waste (Rosenberger, et al., 2005). DOE expects Se-79 and I-129 to dominate the dose to a hypothetical receptor (under 10 CFR 61.41); these radionuclides are expected to contribute only 0.003 percent and 0.001 percent of the radioactivity in the SDF (Rosenberger, et al., 2005). If saltstone oxidation progresses more rapidly than DOE expects, Tc-99, which is expected to contribute approximately 1 percent of the radioactivity in the SDF, is expected to dominate the dose to a hypothetical receptor (under 10 CFR 61.41). Because large variations in the inventories of any radionuclide other than Cs-137/Ba-137m will not significantly affect the total radioactivity in the SDF, information about the total amount of radioactivity in the SDF will not be a sufficient basis to conclude whether the SDF inventory is consistent with meeting the performance objectives. Thus the reviewer must evaluate inventories of individual radionuclides rather than simply the total radioactivity sent to the SDF.

NRC staff based its TER conclusions on DOE's projected SDF inventory and concentrations (see Tables 2 and 3 of this plan). NRC staff based its conclusions on DOE's projected SDF inventory and concentrations rather than DOE's SDF inventory limits because, as discussed in the TER, NRC staff concluded that the deterministic basecase analysis DOE used to develop SDF inventory limits was not demonstrably conservative. Therefore, the reviewer should compare information about the SDF inventory and radionuclide concentrations to the values in Tables 2 and 3 of this plan rather than the SDF inventory limits DOE presented in the PODD and developed in the 2005 special analysis (Cook, et al., 2005).

If DOE proposes new concentration or inventory limits and the NRC staff determines that the limits are technically justified, compliance may be monitored by comparison to the new limits instead of by comparison to the projected inventories or concentrations used to support the waste determination. To evaluate the basis for any new inventory or concentration limits, the NRC staff should evaluate the PA analysis supporting the proposed limits according to the PA guidance in the draft Standard Review Plan for Activities Related to U.S. Department of Energy Waste Determinations (SRP) (NRC, 2006).

In addition to the total inventory, the concentrations of radionuclides in saltstone are expected to affect the potential dose to a hypothetical receptor. DOE expects the primary radionuclide release mechanism from saltstone will be diffusion out of the saltstone matrix. Because the diffusive flux of a radionuclide is proportional to its aqueous concentration gradient, in a diffusive release model, the flux of a radionuclide out of the matrix depends on its pore water concentration. In DOE's PA, pore water concentrations have been modeled with a sorption model, so modeled pore water concentrations are proportional to modeled concentrations in the solid. Thus in DOE's model, radionuclide concentrations in the saltstone solid are expected to directly affect predicted radionuclide release. Radionuclide pore water concentrations that are limited by solubility rather than sorption would not be directly proportional to the concentrations in the saltstone solid. However, as documented in its TER, NRC staff found the sorption limited release model that DOE used to support its waste determination appropriate. For this reason, the NRC staff's conclusion that the waste in the SDF would meet the performance objectives of 10 CFR Part 61, Subpart C, is based, in part, on the assumption that the concentrations of highly radioactive radionuclides (HRRs) in the waste would not be greater than the projected

concentrations that were used to support DOE's waste determination. To verify this assumption, the NRC staff should request information about the concentrations of radionuclides in each batch of waste being sent to saltstone.

The NRC staff is expected to monitor radionuclide inventories and concentrations until the SDF is closed. Unless DOE proposes new inventory or concentration limits that are found to be technically justified by the reviewer, compliance is expected to be based on staff verifying that SDF radionuclide inventories are not likely to exceed the inventories used to support the waste determination (Table 1) and that radionuclide concentrations do not exceed the projected concentrations used to support the waste determination (Table 2).

NRC staff should make two types of comparisons with the inventory data. First, to evaluate data traceability and reproducibility, the reviewer should track inventory data over time and verify that the reported inventory is consistent with previous reported inventories and the activity of individual radionuclides added to the SDF each year. Second, the reviewer should compare the inventory data to the inventory data used to support the waste determination. If, based on the rate of radionuclide additions to the SDF, the reviewer expects that the final inventory of a HRR will exceed the inventory used in the analyses supporting the waste determination, NRC staff should determine whether the increase in inventory is likely to significantly affect the dose to a hypothetical receptor. If so, staff should ask DOE to (i) support the conclusion that the final inventory is not expected to exceed the value used to support the waste determination or (ii) to provide a new analysis that demonstrates that the SDF will meet the performance objectives with the new projected inventory. The compliance demonstration method is chosen by DOE and is not limited to the development of new inventory limits and the use of a sum-of-fractions approach. However, as previously discussed, the reviewer should ensure that a new compliance demonstration does not rely on the inventory limits established in the 2005 special analysis (Cook, et al., 2005). This distinction is important because NRC staff based its conclusions on the projected SDF inventory rather than the SDF inventory limits, which, for several highly radioactive radionuclides, are several orders of magnitude greater than the projected inventories.

DOE intends to separate and treat salt waste with a two-phase, three-part approach (DOE, 2005a). The three processes include (i) processing low-curie, low-actinide waste with deliquification, dissolution, and adjustment (DDA); (ii) processing low-curie, high-actinide waste with a combination of DDA, an actinide removal process (ARP), and a modular caustic side solvent extraction unit (MCU); and (iii) processing high-curie, high-actinide waste with the salt waste processing facility (SWPF). To determine whether the final inventory of a particular radionuclide appears to be likely to exceed the value used to support the waste determination, NRC staff should monitor increases in radionuclide inventories in the context of DOE's salt waste disposal plan. For example, according to information used to support the waste determination, approximately 1 percent of the technetium in the projected final SDF inventory is expected to be added to the SDF during DDA treatment, and only an additional 1.4 percent is expected to be added during ARP/MCU treatment. Thus if staff found that approximately 10 percent of the final projected inventory of technetium was added to the SDF during the initial stage of treatment, it could indicate that the concentration of technetium in the salt waste was much greater than expected. In that case, staff should ask DOE to explain whether the projected inventory of technetium was likely to be exceeded and, if so, how the performance objectives would be met with the new projected inventory and concentration of technetium in the waste.

Table 1. 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	$\mathbf{2.74 \times 10^{6}}$

*Table 3-2: Rosenberger, K.H., B.C. Rogers, and R.K. Cauthen. "Saltstone Performance Objective Demonstration Document." CBU–PIT–2005–00146. Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2005.

Table 2. Projected Radionuclide Concentrations in Grouted Salt Waste To Be Sent to the Saltstone Disposal Facility for Each State of Treatment*				
Ci/gal	**DDA†**	**ARP‡/MCU§**	**SWPF‖**	**Total**
H-3	2.33E−04	2.03E−04	6.91E−05	8.68E−05
C-14	6.94E −06	6.05E−06	4.56E−06	4.80E−06
Co-60	4.96E−06	2.09E−06	6.00E−07	1.01E−06
Ni-59	4.78E−08	1.17E−08	2.44E−08	2.61E−08
Ni-63	1.02E−05	5.06E−06	1.47E−06	2.31E−06
Se-79	1.11E−07	4.32E−07	9.06E−07	8.25E−07
Sr-90	6.14E−04	1.35E−04	1.39E−05	6.88E−05
Y-90	6.14E−04	1.35E−04	1.39E−05	6.88E−05
Nb-94	3.38E−12	3.40E−12	7.12E−12	6.70E−12
Tc-99	3.82E−05	1.60E−04	3.36E−04	3.06E−04
Ru-106	2.71E−06	1.10E−05	2.32E−05	2.11E−05
Rh-106	2.71E−06	1.10E−05	2.32E−05	2.11E−05
Sb-125	1.24E−05	4.48E−05	9.39E−05	8.56E−05
Sn-126	5.30E−07	2.18E−06	4.58E−06	4.17E−06
I-129	2.21E−08	6.62E−08	1.81E−07	1.65E−07
Cs-134	2.63E−04	9.20E−05	5.79E−08	2.51E−05
Cs-135	4.53E−07	1.56E−07	9.84E−11	4.33E−08
Cs-137	1.31E−01	4.58E−02	2.89E−05	1.25E−02
Ba-137m	1.24E−01	4.34E−02	2.73E−05	1.18E−02
Ce-144	4.44E−08	2.88E−08	6.03E−08	5.81E−08
Pr-144	4.44E−08	2.88E−08	6.03E−08	5.81E−08
Pm-147	3.73E−05	1.86E−05	3.90E−05	3.83E−05
Eu-154	8.21E−06	4.41E−06	9.24E−06	9.03E−06
Th-232	7.01E−13	9.87E−14	1.08E−09	9.63E−10
U-232	1.07E−11	1.97E−15	2.22E−10	1.98E−10
U-233	1.06E−07	3.11E−09	1.28E−08	2.06E−08
U-234	3.30E−07	3.32E−09	8.70E−09	3.62E−08
U-235	6.62E−10	5.27E−11	6.00E−10	5.91E−10
U-236	1.53E−08	6.09E−10	1.65E−09	2.80E−09
U-238	6.85E−09	3.13E−10	5.22E−08	4.70E−08
Np-237	6.17E−08	2.76E−09	1.59E−08	1.95E−08
Pu-238	4.06E−04	1.66E−05	1.02E−04	1.26E−04
Pu-239	4.76E−06	2.44E−07	6.36E−06	6.06E−06
Pu-240	1.16E−06	8.3E−08	1.71E−06	1.62E−06

Table 2. Projected Radionuclide Concentrations in Grouted Salt Waste To Be Sent to the Saltstone Disposal Facility for Each State of Treatment* (continued)				
Ci/gal	**DDA†**	**ARP‡/MCU§**	**SWPF‖**	**Total**
Pu-241	2.45E−05	1.46E−06	7.1E−05	6.51E−05
Pu-242	1.9E−10	9.3E−12	1.78E−09	1.6E−09
Am-241	1.39E−06	1.76E−06	8.04E−07	8.8E−07
Am-242m	5.79E−10	1.02E−09	4.63E−10	4.88E−10
Cm-244	1.72E−07	7.06E−07	8.71E−07	8.07E−07
Cm-245	1.68E−11	6.95E−11	8.58E−11	7.94E−11
Na-22	2.57E−05	3.01E−05	4.93E−05	4.68E−05
Al-26	9.66E−08	1.34E−07	2.32E−07	2.18E−07
Te-125m	3.02E−06	1.09E−05	2.29E−05	2.09E−05
Sb-126	7.42E−08	3.06E−07	6.41E−07	5.84E−07
Sb-126m	5.3E−07	2.18E−06	4.58E−06	4.17E−06
Sm-151	3.23E−05	2.09E−05	4.37E−05	4.21E−05
Eu-152	1.56E−07	1.01E−07	2.11E−07	2.04E−07
Eu-155	1.82E−06	1.18E−06	2.47E−06	2.38E−06
Ra-226	2.63E−08	7.49E−15	1.33E−07	1.2E−07
Ra-228	7.01E−13	9.87E−14	1.08E−09	9.63E−10
Ac-227	1.51E−13	1.2E−14	1.85E−13	1.77E−13
Th-229	3.01E−10	8.86E−12	4.91E−11	6.98E−11
Th-230	1.6E−10	9.16E−13	3.53E−10	3.27E−10
Pa-231	4.19E−13	3.34E−14	5.13E−13	4.92E−13
Pu-244	1.24E−12	4.25E−14	8.19E−12	7.38E−12
Am-243	5.22E−10	3.37E−10	1.54E−10	1.9E−10
Cm-242	4.79E−10	8.33E−10	1.03E−09	9.76E−10
Cm-243	3.06E−10	1.98E−10	2.43E−10	2.48E−10
Cm-247	5.89E−20	3.8E−20	4.69E−20	4.77E−20
Cm-248	6.13E−20	3.96E−20	4.89E−20	4.97E−20
Bk-249	4.48E−27	2.9E−27	6.07E−27	5.85E−27
Cf-249	3.4E−19	2.2E−19	4.61E−19	4.44E−19
Cf-251	1.16E−20	7.52E−21	1.58E−20	1.52E−20
Cf-252	3.78E−22	2.44E−22	5.11E−22	4.93E−22

*d'Entremont, P.D. and M.D. Drumm. "Radionuclide Concentrations in Saltstone." CBU–PIT–2005–00013. Rev. 3. Aiken, South Carolina: Westinghouse Savannah River Company. 2005.
†DDA = Deliquification, dissolution, and adjustment
‡ARP = Actinide removal process
§MCU = Modular caustic side solvent extraction unit
‖SWPF = Salt waste processing facility

As previously discussed, staff are expected to compare radionuclide inventories to the inventories used to support the waste determination unless DOE develops and NRC reviews changes to the PA that account for the new inventories. Thus to determine whether the inventory of a particular radionuclide appears to be likely to exceed the value used to support the waste determination, staff should use the radionuclide concentrations and waste volumes DOE planned to send to the SDF at the time of the waste determination. Radionuclide concentrations that DOE expects in waste from each treatment phase are given in Table 2. Both DOE's draft and final waste determinations indicate that DOE plans to add 32,600 m^3 [8.6 million gal] of DDA waste, 10,600 m^3 [2.8 million gal] of APR/MCU waste, and 363,000 m^3 [95.8 million gal] of SWPF waste to the SDF. However, NRC staff expects that the volume of waste to be treated with each process will be different from the volumes presented in the waste determination. For example, a recent draft modification of SCDHEC's solid industrial waste landfill permit for the SDF (SCDHEC Permit #025500-1603) would limit the amount of waste treated by DDA alone to approximately 4,660 m^3 [1.23 million gal].

NRC staff should make two types of comparisons with the concentration data. First, to evaluate data traceability and reproducibility, staff should verify that the reported radionuclide concentrations and volumes of batches of waste sent to saltstone are consistent with the reported inventories of radionuclides in the SDF. Second, staff should compare the concentrations of radionuclides in each batch of waste sent to the SDF and determine whether they exceed the projected concentrations used to support the waste determination. In making both of these comparisons, staff should account for the uncertainty in the measured radionuclide concentrations and volumes of waste. This part of the review should be coordinated with the review of feed tank sampling (Section 3.1.6).

Because the characteristics of salt waste in each tank and the characteristics of the different treatment processes DOE plans to use to treat the salt waste differ, the concentrations of radionuclides in waste sent to saltstone are expected to vary from batch to batch. One instance in which the concentration exceeds the expected value is not expected to have a significant effect on the predicted dose to a hypothetical receptor, unless the concentration is so different from the expected value that it significantly changes the expected radionuclide inventory. However, if a radionuclide concentration consistently exceeds the concentration used to support the determination, staff should determine whether the change in concentration or the resulting change in inventory is likely to significantly affect the dose to a hypothetical receptor. If so, staff should ask DOE to demonstrate that the SDF will meet the performance objectives with the new projected radionuclide inventories or concentrations.

Although a risk-informed review is expected to focus on the inventories and concentrations of the radionuclides that were identified as HRRs in the saltstone review (i.e., Cs-137/Ba-137m, Sr-90/Y-90, Pu-238, Am-241, Cm-244, Pu-239, Se-79, I-129, and Tc-99), staff should also note any large deviations of the inventories and concentrations of other radionuclides from the expected values. Large deviations may be important because the selection of HRRs was based on the inventories in Table 1 and concentrations in Table 2. Radionuclides that were not identified as HRRs in the salt waste determination are not expected to significantly affect the potential dose to members of the public. However, if the concentrations or inventories of a radionuclide not identified as an HRR is significantly greater than expected (e.g., by a factor of 5 or greater), staff should use the information in the PODD and TER, or in any revised PA analysis provided by DOE, to determine whether the deviation could significantly affect the

predicted dose. If it does, staff should ask DOE to explain how the waste will meet the performance objectives with the new projected radionuclide inventory or concentration.

3.1.1.2 Treatment Efficiencies

As described in DOE's final waste determination (DOE, 2006), DOE intends to separate and treat salt waste with three separate treatment processes, including (i) DDA for low-curie and low-actinide waste; (ii) a combination of DDA, ARP, and MCU for low-curie and high-actinide waste; and (iii) the SWPF for high-curie and high-actinide waste. The radionuclide removal efficiencies achieved by each of these processes affect the predicted doses to hypothetical receptors because they affect the inventory of the disposal facility. In addition, the expected treatment efficiencies form part of the basis for demonstrating that HRRs have been removed to the maximum extent practical. Because the removal of radionuclides to the maximum extent practical served as the basis for concluding that doses would be maintained ALARA, significant deviations from the predicted removal efficiencies will affect DOE's demonstration that the ALARA requirement of 10 CFR 61.41 is satisfied. For these reasons, the NRC staff identified radionuclide removal efficiencies as a key factor to be monitored in assessing compliance with the performance objectives.

Monitoring activities related to radionuclide removal efficiencies are expected to continue until salt waste treatment is completed and waste is no longer being transferred to the SDF. Monitoring of radionuclide removal is expected to include two types of activities. The first, described in this section, involves reviewing removal efficiency data DOE report to evaluate data traceability, reproducibility, and representativeness. The second monitoring activity, which is discussed in Section 3.1.8, focuses on comparing the actual removal efficiencies achieved to the projected removal efficiencies used to support the waste determination.

To evaluate data traceability and reproducibility, NRC staff should request information about the actual treatment efficiencies achieved, as well as information about radionuclide concentrations in the inputs to and effluents from the various treatment processes. Staff should then determine whether the radionuclide concentrations are consistent with reported treatment efficiencies. If the staff's calculated efficiencies are different from DOE's reported efficiencies, staff should ask DOE to demonstrate their calculation and explain the differences. In addition, staff should verify that reported removal efficiencies and information about process inputs and outputs are consistent with the reported inventory additions to the SDF. Specifically, given the radionuclide concentrations and volume of a batch of waste and the reported radionuclide removal efficiencies, the staff should calculate the expected increases in radionuclide inventories due to each batch of waste. Staff should coordinate this review with the review of radionuclide inventory (Section 3.1.1.1) to verify that the expected additions to the SDF based on radionuclide removal efficiencies are consistent with the reported SDF inventory.

To the extent practical, staff should also evaluate available information about subprocesses and evaluate consistency with treatment process efficiency. For example, the primary radionuclide removal mechanisms of the DDA process are the removal of dissolved radionuclides by deliquification of the salt waste and removal of particle-associated radionuclides by settling. If information about these mechanisms is available, staff should verify that the efficiencies are consistent with the reported removal of dissolved and particle-associated radionuclides, respectively.

To evaluate the representativeness of reported treatment efficiencies, the staff should note any significant differences in reported removal efficiencies from those that have been previously reported. If possible, the staff should determine whether the change in efficiency is due to changes in waste characteristics or changes in the treatment process. The staff also should note any observed trends in treatment efficiencies. As part of the monitoring activities related to Factor 8, Removal Efficiencies, the staff will determine whether any observed trends are likely to affect meeting the performance objectives (Section 3.1.8).

3.1.1.3 Environmental Data

Review of the environmental data DOE collects is a necessary part of the monitoring to assess compliance with the requirements for the protection of the general public in 10 CFR 61.41 because it will provide insights on SDF performance and the potential for the public to receive a dose from the waste disposed there. For example, the presence of increased amounts of radionuclides in environmental samples, such as in the groundwater samples obtained near the SDF, may indicate that the radionuclides are leaching from the vaults and that the performance of the facility is less than expected.

DOE conducts an effluent monitoring and environmental surveillance program on an ongoing basis at the SRS. The data obtained through this program are summarized in an annual environmental report. As described in more detail in Section 1.3, the State of South Carolina also requires that DOE monitor the groundwater at the SDF as part of their solid waste permit. The State of South Carolina also collects environmental data at SRS and in surrounding areas through ESOP. A variety of environmental media including groundwater, surface water, rainwater, air, vegetation, deer and hog meat, and soil are monitored through these programs. The environmental data that will be the most useful to monitor to assess compliance with 10 CFR 61.41 is the groundwater data from the Z-Area because it is expected that the predominant pathway for long-term releases from the disposal facility will be the groundwater pathway. It may be useful to also check the other environmental data collected at SRS to verify that there are no substantial increases in contamination. However, the usefulness of this data is somewhat limited because most of these samples are not obtained directly in the vicinity of the SDF, and there are other potential sources of radioactivity at SRS, making it difficult to determine whether any observed concentration increases are attributable to the waste disposed at the SDF.

The current groundwater monitoring plan for the SRS Z-Area is described in the Groundwater Monitoring Plan for the Z-Area Saltstone Disposal Facility (DOE, 2005b). According to this plan, DOE plans to monitor the groundwater in three water table wells located downgradient of Vault 4 (designated ZBG3, ZBG4, and ZBG5) as well as in one well located upgradient of the SDF (designated ZBG1). In addition, three new wells (designated ZBG6, ZBG7, and ZBG8) have been recently installed downgradient of Vault 1. In the future, if new vaults are constructed downgradient of the current wells, new wells will be installed within 30 m [100 ft] downgradient of the new vaults. Under the current groundwater monitoring plan, groundwater samples are taken on a semiannual basis and are analyzed for nitrate/nitrate, gross alpha, gross beta, I-129, and tritium. The measurement of nitrate is important because it is very mobile and is present in high concentrations in the saltstone and can therefore serve as an early indicator of leaching from the SDF.

At the onset of monitoring, NRC staff should request updated information about the groundwater monitoring plan for the SDF to determine whether there have been any changes from the 2005 plan. NRC staff should evaluate the groundwater monitoring plan to verify that the sampling frequency is adequate and that a sufficient number of wells are installed in appropriate locations. DOE's groundwater monitoring plan should also be evaluated to determine whether their list of monitoring constituents includes all relevant constituents. With the exception of tritium and I-129, the current groundwater monitoring plan includes the measurement of gross alpha and beta instead of individual radionuclides. NRC staff should verify that these analytical methods are sensitive enough to detect low levels of radionuclides leaching from the SDF. It may be useful to coordinate the review of DOE's groundwater monitoring plan with the State because the State permits have specific groundwater monitoring requirements.

NRC staff should also request DOE's quality assurance plans for acquiring and measuring environmental samples at the onset of monitoring. Staff should evaluate the quality assurance program to determine whether the program will provide adequate assurance that the samples are collected and analyzed properly. This program should include a mechanism for verifying that the laboratory analysis is accurate, such as the use of spike samples or duplicate samples, as well as a mechanism for verifying that contamination of the samples is not occurring, such as the use of blank samples. In addition, the Quality Assurance and Quality Confirmation program should include appropriate record keeping and chain of custody procedures.

Each year, NRC staff should request the groundwater monitoring data DOE obtains. This data should include the measured concentrations in the groundwater along with the uncertainty associated with these measurements and the analytical detection limits for the methods used to obtain the measurements. At the beginning of monitoring, NRC staff should also request the historical concentrations of the radionuclides and nitrate that were measured in the groundwater near the SDF. This data is needed to establish baseline levels at this site because some of the analytes of interest are naturally present, and there are other potential sources of contamination at SRS. Without adequately knowing background levels in the Z-Area, it may be difficult to determine whether any increases in measured radioactivity or nitrate are due to leaching from saltstone. NRC staff should also consider measured concentrations in the upgradient well when establishing background levels. The concentrations measured in the groundwater should be tracked over time to determine whether there are any trends in the data. Any observations of increased radioactivity levels or of nitrate should be followed up with additional sampling and analysis, and the source of these increased levels should be determined. If the contamination shows that there is more leaching from the SDF than expected, the reason for this should be determined. DOE should also update its PA to reflect the observed performance and to determine whether the performance objectives can still be met.

The review of environmental data to assess compliance with 10 CFR 61.41 will be an open issue at the onset of monitoring and will remain open indefinitely. The environmental monitoring data should be reviewed annually, and the conclusions should be included in the annual monitoring report. It is anticipated that the initial reviews of DOE's Z-Area groundwater monitoring plan and quality assurance plans will be extensive, but that subsequent reviews will be much less in depth if the initial review concludes that these plans are adequate. The groundwater monitoring plan may need to be reevaluated by NRC staff as new vaults and wells are constructed or if there is evidence that the performance of the SDF is worse than expected (e.g., environmental data shows that leaching from the saltstone is occurring, extensive

cracking of the wasteform is observed).

3.1.2 Factor 1—Oxidation of Saltstone

NRC's technical evaluation of the saltstone disposal facility (NRC, 2005a) indicated that saltstone oxidation was expected to be a key factor in determining whether the facility could meet the performance objective of 10 CFR 61.41. In particular, the NRC TER noted the following:

> "The rate of waste oxidation and release of technetium from an oxidized layer of saltstone will be a key determination 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."

Thus saltstone oxidation is considered to be important to meeting the performance objective primarily because oxidation can lead to increased releases of technetium from the wasteform. In the TER, the NRC staff based its conclusion that there was reasonable assurance that the performance objectives in 10 CFR 61.41 would be met on several assumptions, including the assumption that DOE would perform more realistic modeling of waste oxidation and that the results would demonstrate that release rates of technetium will be acceptable (NRC, 2005a). To validate this assumption, NRC staff expect to review models of the predicted oxidation of saltstone and resulting release of technetium as well as experiments designed to provide support for the models. Factors 2 and 3 identify aspects of saltstone oxidation and radionuclide release that require additional model support, including cracking of saltstone and flow through fractures in the wasteform. In addition, Factor 3 indicates that support for the results of oxidation models, such as observations of oxidation in saltstone or surrogate wasteforms, are needed. Thus monitoring of Factor 1 should be coordinated with the reviews of Factors 2 and 3.

The dry grout formula that DOE plans to use to make saltstone contains blast furnace slag that creates chemically reducing conditions to immobilize technetium. Tc(VII) is believed to react with sulfides from the slag to form Tc_3S_{10} (Lukens, et al., 2005). Retention of technetium in grout made with blast furnace slag is supported by the results of DOE field lysimeter experiments that have shown that technetium release from wasteform samples that contain slag is significantly slower than technetium release from samples that do not contain slag (McIntyre and Wilhite, 1987). In its 1992 PA, DOE used geochemical analyses to derive the initial pore solution concentrations of Tc-99, I-129, and nitrate (Cook and Fowler, 1992). However, in the 2005 special analysis (Cook, et al., 2005) that supported the waste determination, DOE instead modeled Tc release with a linear partitioning coefficient (K_d) and used a literature value for the K_d of Tc in reducing concrete (Bradbury and Sarott, 1995).

Because the K_d value used in DOE's basecase model was applied to the entire wasteform and did not change with time, DOE's basecase results rely on the assumption that the entire wasteform will remain reducing for the duration of the performance period. However, some

oxidation is expected to occur during the performance period. In response to NRC questions about saltstone oxidation, DOE indicated that they expected 3 to 8 percent of the reducing grout in a grouted high-level waste tank to oxidize during the 10,000-year performance period (WSRC, 2005a). Oxidation is expected to occur at saltstone surfaces, including the surfaces of cracks, where the saltstone is exposed to water or, potentially, to air. NRC staff believe that an oxidized layer at saltstone surfaces could significantly increase technetium release. In DOE's basecase model, the primary mechanism for release of radionuclides from saltstone is diffusion. Because diffusive transport is directly proportional to the concentration gradient in the fluid phase and fluid phase concentrations of technetium are many orders of magnitude higher under oxidizing conditions than they are under reducing conditions, releases from saltstone are expected to be sensitive to the rate of formation of the oxidized layer. Furthermore, an oxidized layer could affect technetium release significantly if the infiltration cap degrades more rapidly than assumed in DOE's basecase model, because in an oxidized layer more mobile forms of technetium would coincide with pathways for water through the waste.

DOE's sensitivity analyses demonstrated that if all other system components performed as expected, complete oxidation of saltstone would result in potential doses close to the dose limit (NRC, 2005a). DOE's sensitivity analyses also showed that unacceptable releases of technetium could occur if the saltstone became completely oxidized and hydraulic performance also deteriorated. For example, unacceptable releases were predicted to occur if the saltstone was completely oxidized and if the waste became completely saturated, allowing flow in fractures. Similarly, the sensitivity analyses indicated that much higher doses could occur if complete oxidation took place, precipitation increased by 25 percent, and the hydraulic conductivity of the vault and saltstone degraded more than expected during the performance period (NRC, 2005a). Because some oxidation is expected but complete oxidation could lead to unacceptable results in several plausible scenarios, more realistic models that simulate partial oxidation of saltstone during the performance period are needed to demonstrate that the performance objectives will be met.

Additional model assessment guidance, including evaluation of technical model features such as grid size and the size of timesteps, is provided in Section 3.4 of NRC's draft Standard Review Plan for Activities Related to U.S. Department of Energy Waste Determinations (NRC, 2006). In addition to these general review areas, there are two specific issues that staff should review with respect to a saltstone oxidation model.

The first specific review area is the consistency of modeling results with saltstone surfaces that could be exposed to water or air. Because oxidation is expected to proceed from saltstone surfaces, the model should be consistent with the geometry of the saltstone wasteforms and the size and frequency of cracks expected to develop in the wasteforms during the performance period. In the waste determination and supporting documents, DOE proposed to create the SDF from approximately 14 monolithic rectangular blocks of saltstone, poured directly into concrete vaults. Changes in this design that increase the surface area-to-volume ratio of the saltstone blocks are expected to increase saltstone oxidation and, therefore, the release of Tc. In addition, changes in the design that introduce pathways for water through the waste, such as the addition of steel support beams inside vaults, introduce additional potential for cracking, infiltration of water through the waste, and consequently, additional oxidation of saltstone. Because of the importance of the wasteform geometry and disposal site construction to saltstone oxidation, this part of the review should be coordinated with the onsite observations of vault construction (Section 3.2.4). Additional characterization of the size and frequency of

cracks that are expected to develop in saltstone during the performance period is expected to be developed to support Factors 2 and 3 (Section 3.1.3 and 3.1.4).

The second specific review area is gas phase transport of oxygen in saltstone. DOE's predicted oxidation of grout in high-level waste tanks was based on the assumption that oxygen was transported in the grout only as oxygen dissolved in water. In response to NRC questions about the potential effect of gas phase transport on saltstone oxidation, DOE replied that oxygen in soil gas could increase the amount of saltstone that is oxidized, but that the process was believed to be slow because of the slow diffusion of oxygenated water (WSRC, 2005a). This response appears to be based on the assumption that gas phase oxygen would not penetrate the saltstone wasteform. However, DOE has not shown that small cracks in saltstone, which were not included in DOE's analysis of saltstone oxidation, would remain saturated. Furthermore, the assumption that the saltstone matrix will remain 100 percent saturated is inconsistent with the use of a relative permeability of the saltstone matrix of less than unity in the basecase analysis (Cook, et al., 2005). Unless DOE demonstrates that gas phase transport within the saltstone matrix and fractures (including small fractures) does not significantly affect saltstone oxidation, NRC staff should review how the oxidation model accounts for gas phase transport of oxygen within the saltstone matrix and fractures. Staff also should ensure that wasteform saturation assumptions are supported and are consistent with assumptions made in other parts of the model (e.g., values of relative permeability of saltstone).

Closure of this key factor is expected to depend on the results of field and laboratory experiments as well as improved modeling of saltstone oxidation and technetium release. The need for oxidation model support is identified in Factor 3. Such support may include measurements of the depth of penetration of the oxidation front in laboratory samples exposed to oxygenated water or, if appropriate, water and air. In that case, NRC staff should pay particular attention to the validity of extrapolations to the larger saltstone wasteform as well as the longer performance period. In addition, field experiments may provide direct evidence of both the rate of oxidation expected for saltstone samples buried at the SDF and the resulting release of technetium. As mentioned previously, DOE installed field-scale saltstone lysimeter tests with and without slag and found that there was a much greater release of technetium from the lysimeter made without slag than from the lysimeter with slag (Cook and Fowler, 1992). As described in the TER, this data also showed an increasing technetium release starting around day 250, which was consistent with the predictions of a PORFLOW (Analytical and Computational Research, Inc., 2002) simulation (WSRC, 2005a). Because increased technetium releases were observed at the end of the lysimeter tests, longer term field scale results would help support the predicted technetium release rates. In addition to providing information on field-scale release rates, the slag lysimeters may contain valuable information about the rate of saltstone oxidation. In the analyses used to support the waste determination, DOE's estimates for the amount of saltstone oxidation over 10,000 years are based primarily on numerical modeling results. If the lysimeter made with slag was exhumed, it may be possible to measure the depth of penetration of the oxidation front to support the rate of saltstone oxidation under field conditions.

In addition to direct support for the extent of oxidation and technetium release, support is needed for aspects of the oxidation and release models, including predicted wasteform cracking, saturation of the wasteform, and moisture flow through fractures. The need for model support for wasteform cracking is identified in Factors 2 and 3, and the need for support for moisture flow through fractures is identified in Factor 3. Thus reviews of model support for

these aspects of the oxidation model should be coordinated with the reviews of Factors 2 and 3. Support for the expected degree of cracking of the saltstone wasteform during the performance period may include examination of saltstone samples or surrogate wasteforms buried at the site; laboratory tests of accelerated aging of grout samples; and process-level modeling of cracking due to shrinkage, expansive phase formation, settling, or other mechanisms. In general, support for a PA model should include the results of physical experiments and should not rely solely on process-level models. In addition to the results of field and laboratory experiments, NRC staff should be aware of any observations of cracking in saltstone vaults or wasteforms that are made during onsite observations (Section 3.2).

As previously discussed, information about wasteform saturation is needed to predict the relative permeability of the wasteform as well as the potential for gas phase movement of oxygen in the wasteform. As explained in DOE's response to NRC's RAI (WSRC, 2005b), the effective permeability results for saltstone used in DOE's POREFLOW analyses were not regarded as reliable because it appeared that the saltstone sample or testing configuration was damaged during one of the measurements. Thus, if measured rather than bounding values are used in future revisions of the PA, experiments supporting the predicted extent of wasteform saturation should include new measurements of wasteform moisture characteristic curves. In assessing potential process-level models of wasteform saturation and flow through fractures, staff should also consider (i) heterogeneity in material properties; (ii) temporal variations in saturation, especially resulting from rapid transport of infiltration through root holes to the lower drainage layer; (iii) infilling of fractures and joints with porous media; (iv) offset of the vault or saltstone on either side of a fracture; (v) variability in apertures of joints and fractures; (vi) sensitivity to grid size in simulations, especially at the interface of a fracture and porous media; and (vii) variability in moisture characteristic curve parameters.

The review of information related to saltstone oxidation is expected to continue until sufficient model support shows that the predicted saltstone oxidation is likely to minimally affect dose. Specifically, the review is expected to continue until (i) sufficient model support is available to limit the uncertainty in the predicted extent of oxidation and resulting technetium release such that the predicted dose meets the performance objectives or (ii) DOE demonstrates that oxidation does not significantly affect the predicted dose to a hypothetical member of the public. Based on the available information about technetium chemistry, it is not expected that (ii) will be shown.

3.1.3 Factor 2—Hydraulic Isolation of Saltstone

The NRC TER (NRC, 2005a) concluded that the following factor is important to assessing whether DOE's disposal actions will be compliant with the performance objectives of 10 CFR 61.41:

> "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 [toxicity characteristic leaching procedure] 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."

Because the performance of the wasteform and concrete vault barriers is important to meet the performance objectives in 10 CFR Part 61.41, the methods used to analyze the long-term performance of the saltstone disposal facility must account for potential mechanisms of contaminant release from the facility and the potential mechanisms for loss of integrity of the saltstone disposal facility engineered barriers. As described in the TER, DOE's demonstration that the disposal system could meet the performance objectives was largely determined by several attributes, including (i) water passing through the saltstone matrix is limited, (ii) radionuclide diffusion rates are limited, (iii) water passing through fractures is limited, and (iv) oxidation near saltstone boundaries is limited. DOE's sensitivity analyses of Vault 4 performance suggest water flow rates, degradation rates of the wasteform and vaults, and oxidation of the wasteform are the most significant processes. Oxidation near saltstone boundaries is discussed in Section 3.1.2. Technical review monitoring activities related to water flow through the saltstone matrix, radionuclide diffusion, and water flow through fractures are discussed in this section.

Limiting the amount of water flow through the wasteform is important to achieving 10 CFR 61.41. 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 wasteforms through chemical (e.g., sulfate attack, carbonation, leaching, and rebar corrosion) or physical (e.g., cracking due to settlement of foundation soil, shrinkage, seismic activity) processes is likely to increase their hydraulic conductivity and therefore increase the contaminant release. Based on updated low hydraulic and effective diffusivity properties of the initial materials, it may be acceptable to assume that degradation mechanisms will not result in an increase of the bulk hydraulic conductivity of the wasteform to greater than 1×10^{-7} cm/s [1×10^{-1} ft/yr] 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 DOE studies to justify the assumed long-term degradation rates of the wasteform may significantly reduce uncertainty.

Degradation that alters the integrity and permeability of the saltstone wasteforms and vaults was considered in the 2005 special analysis (Cook, et al., 2005). Cracks in the saltstone were not included in the analysis. 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. The hydraulic conductivities of saltstone and concrete in the time interval of 0 to 100 years were set to 1×10^{-11} and 1×10^{-12} cm/s [1×10^{-5} and 1×10^{-6} ft/yr], respectively, and were increased to 1×10^{-9} cm/s [1×10^{-3} ft/yr] over a 10,000-year period through 8 steady-state stages. These assumptions were based on

professional judgment because actual data over the time periods of interest do not exist. For comparison, values of hydraulic conductivities of concrete and cement pastes available in the literature were tabulated by DOE (WSRC, 2005a). For example, DOE reported hydraulic conductivity measurements by Malek, et al. (1985) that ranged from 2.5×10^{-6} cm/s to $< 1.0 \times 10^{-11}$ cm/s [2.5 ft/yr to 1.0×10^{-5} ft/yr] for nondegraded samples made with earlier saltstone formulations. However, whether the variability in the measurements is a function of curing time or the variability results from the measurement technique was uncertain. 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 the uncertainties. In the current design, the vaults contain rebar and fill pipes that would be exposed to the environment. Unless mitigated by future design modifications, the effects of rebar and fill pipe corrosion on the integrity and projected lifetime of the vaults in future degradation calculations should be considered.

Cracking would be expected to significantly shorten the diffusive path length and could increase the effective diffusion coefficient depending on the frequency and severity of the cracking. Because 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. Because of uncertainties associated with the expected frequency and severity of cracking in the saltstone wasteform and vaults, the TER identified vault and saltstone fracturing as an area that would benefit from additional model support (Factor 3). As discussed in greater detail in Section 3.1.4, model support (i.e., information that provides confidence that the numerical model results adequately represent the behavior of the actual system) can take the form of experimental results, documented analogs, or expert elicitation that substantiates the model assumptions used.

Saltstone grout and vault concrete degradation is represented by increasing hydraulic conductivity of the wasteform or the concrete. These increases are related to increases in porosity of the bulk material that may be caused by leaching, phase changes, and cracking. Chemical and physical degradation mechanisms that could cause changes in the hydraulic conductivity of saltstone grout and the vault concrete include (i) dissolution of salts and low solubility matrix phases; (ii) formation of new or expansive matrix phases due to hydration, precipitation, and recrystallization reactions of the matrix; (iii) formation of new phases as the result of introduction of additional chemical species from the environment; (iv) radiation damage; (v) settling and/or seismic forces; (vi) external static loading; (vii) freeze-thaw cycling; and (viii) erosion of material exposed to the environment.

To evaluate any analysis of the long-term performance of the saltstone vaults and wasteform, NRC reviewers will evaluate which degradation mechanisms were considered and the technical basis for excluding any degradation mechanisms from the analysis. A mechanism may be excluded if it is very unlikely to occur or if its effects are not expected to be significant. For example, if analyses show that grout used to plug 7.6-cm [3-in] diameter fill pipes on the roof of each cell are susceptible to cracking and fracturing, it is possible that the contaminated grout would be unaffected, so the overall significance to the results may still be low. To evaluate potential degradation mechanisms, NRC staff will need to have confidence that all relevant chemical reactions have been considered and that an appropriate range of environmental conditions, including the temperature, which may be dominated by curing and influenced by radiolytic heating, were considered. The chemical inventory should include the material of supporting structures or equipment left behind within the vaults (e.g., carbon steel of pipes).

One degradation mechanism is ettringite formation or sulfate attack, which involves the diffusion of sulfate into the concrete matrix, followed by the reaction of sulfate with components of typical concrete. The reaction products have greater volumes than the reactants, meaning the reaction products expand within the concrete matrix. In instances where the sulfate attack occurs on unconfined concrete surfaces, the reactions lead to expansion and result in the cracking and fracturing of the concrete. If DOE indicates that the resistance of the concrete vaults to sulfate attack is based on a polystyrene sheet drain similar to the one used in Vault 4, then the reviewer should evaluate the technical basis for the assumed lifespan of the sheet drain. Oxidation degradation of the polystyrene is the most likely mechanism of degradation of the sheet drain. If no sheet drain is to be used and DOE indicates that sulfate attack will be slowed by increasing internal pressures and a reduction in the available total porosity (WSRC, 2005b), then the reviewer should evaluate the technical basis supporting DOE's conclusion that increasing internal pressure and reduced porosity will sufficiently protect saltstone from sulfate attack.

Another potential mechanism of vault degradation is cracking caused by hydrostatic pressure. Analyses have indicated that while the cells of Vault 1 were being grouted, excess saltstone cure water, combined with rainwater leaking in and condensate water from the humid atmosphere, provided sufficient hydrostatic pressure on the bottom of the cell walls to cause the vault walls to crack. In addition, Vault 1 has void spaces within the wasteform originating from interior circumference drain pipes and pillars so that reliance on confined conditions within the vault to prevent expansion, spalling, and cracking may be unjustified. Thus, if DOE uses the vault design described in the waste determination, the technical reviewer should evaluate the basis for concluding that the failure that occurred in Vault 1 would not occur in other vaults. If DOE uses an alternate vault design, NRC staff must consider possible failure modes of the alternate design and update the monitoring plan accordingly.

The saltstone wasteform described in DOE's waste determination is a solid product of chemical reactions between a salt solution and a blend of cementitious materials (blast furnace slag, fly ash, and a lime or cement). The recommended composition for saltstone by weight, using a nominal salt solution blend, is 47-percent salt solution, 25-percent slag, 25-percent fly ash, and 3-percent cement. The slag is used to produce reducing conditions that result in low solubilities and increased retardation for certain contaminants. DOE indicated a range of saltstone compositions over which acceptable saltstone can be produced based on results of the toxicity characteristic leaching procedure (TCLP) tests (WSRC, 2005b). 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. An increase in hydraulic conductivity would adversely affect the reducing capacity of the saltstone due to diffusion or advection of oxygen into the wasteform. While the reducing condition is dependent on the amount of slag added to the saltstone–grout mixture, the amount of slag can also affect the amount of shrinkage and cracking during curing. In addition, concrete mix properties also govern the ability of sulfate to diffuse into the concrete mix. Because physical properties such as hydraulic conductivity and effective diffusivity are influenced by the composition of the cementitious material, associated model assumptions must be substantiated. The saltstone grout composition from which the physical properties were determined should be reviewed, as should the justification for the physical properties of saltstone wasteform obtained and the accompanying range of saltstone components. Verification of the grout formula used to make the saltstone wasteform is discussed in Section 3.2.4.

Because workmanship also is expected to affect the hydraulic properties of the concrete and grout, detailed information describing the workmanship should be reviewed. Often, curing techniques are referred to as a component of workmanship (e.g., the workmanship of the Vault 1 construction was documented to be inferior to the workmanship of Vault 4). It has been shown that expansive phases can change within cementitious material as a function of curing time and therefore curing time is significant to waste isolation. Detailed information about the curing techniques and the curing times must be reviewed. Answers to the following questions help understanding: What is the optimal moisture content and temperature during wasteform curing? How long does this environment need to be maintained before the entire wasteform is cured? Is this environment dependent on nature, or can this environment be artificially maintained? What are the curing techniques used and technical basis that it is successful for saltstone? Are the curing time differences between the interior and exterior of the saltstone significant? Will radiolytic heating affect curing? The interrelationship between the components of the cementitious mixture and the curing techniques must be understood because both are important to determining the physical properties of the saltstone grout and vault concrete.

Leaching of soluble materials from saltstone can change the physical properties of the wasteform. For example, nitrate salts make up 13 percent of the grout weight and are expected to leach out at some point in time. Information should be reviewed on the likelihood of a significant amount of water condensing within the vaults and interacting with the wasteform in a significant way such as leaching out of soluble phases. NRC staff should review information on dissolution of salts and low solubility matrix phases to understand their significance and the uncertainties involved and to gain confidence in the model assumptions used.

Besides wasteform and vault cracking due to hydration, precipitation, and recrystallization, cracking from settling and/or seismic forces, external static loading, and freeze–thaw cycling can also occur. As previously discussed, an adequate technical basis must be provided for each fracture-forming degradation mechanism that is excluded from the PA. For example, a technical basis for excluding the freeze–thaw cycling would include the time period when the vaults are exposed to the elements before being covered by the closure cap and protected from the environment. As another example, excluding degradation through seismic activity should have a technical basis that includes the seismic hazard of the area and significance of past earthquakes such as the large 1886 Charleston Earthquake with a magnitude of 7.3. Calculations for external static loading, wasteform structural integrity, and, as previously discussed, the formation of expansive matrix phases, must consider the void spaces within the wasteform originating from interior structures (e.g., circumference drain pipes and pillars made out of carbon steel). The total volume within the vault space that is not filled with cementitious material should be known, and information on its effects on the degradation mechanisms should be reviewed. If relevant, information on the effects of roof and vault wall penetrations on long-term performance should be reviewed. A leachate removal system, which would be designed to remove saltstone cure water and water from other sources such as precipitation and condensation from the vaults, would require drilling holes into the vaults to extract the fluid, while the roof may contain numerous holes to pour the nonradioactive grout evenly on top of the saltstone.

Multiple sources of evidence to support model assumptions about wasteform and vault degradation could include collecting laboratory and field emplaced old and new vault concrete, saltstone grout, and soil and analyzing that material to measure properties including saturated/unsaturated hydraulic conductivity, moisture retention, reducing capacity, diffusion

coefficient, bulk density, and porosity. Experiments also may be designed to study the effects of aging on samples under both field conditions and conditions designed to simulate accelerated aging. It is important that NRC staff analyzes and substantiates extrapolations to the larger saltstone wasteform as well as the longer performance period. If staff can substantiate the extrapolations, the degradation model assumptions could be supported by such laboratory and field studies. Assumptions used in the calculations of flow through fractured vaults and saltstone include the occurrence of vertical fractures with a certain fracture width and a certain fracture spacing. These assumptions, such as the probability of straight vertical fractures versus a branching fracture network as observed in ceramics and concrete, may be substantiated by such information. Branching cracks, along with microcracks that result from mechanical and chemical (e.g., sulfate attack) effects, could lead to higher radionuclide releases compared to vertical fractures. Higher releases also would occur if the fractures were more closely spaced than those assumed in the DOE's PA. Characterization information from Vault 1 could possibly be used because it has experienced cracking. Information from Vault 1 and other sources on vertical versus branching fractures could be documented. Moisture content and potential flow could be analyzed. Measurements and observations could be used to support the important assumption of the cracks pinching out on the top or bottom of the wasteform, halting flow. Field analyses could possibly support current assumptions that shrinkage cracks are filled by the succeeding grout pour so as to preclude fracture flow. If the basis for the assumed degree of cracking in saltstone is observations of cracking of Vault 1, differences between the chemical and physical properties of wasteform and the concrete used in Vault 1 should be addressed. In addition, the ability to observe small cracks and the possible implications of the existence of cracks that are too small to be observed should be reviewed with respect to the hydraulic properties of saltstone as well as saltstone oxidation.

Based on vadose zone simulation by DOE, which indicates the saltstone will experience a suction of roughly 1,200 cm [470 in], while fracture flow would require suction conditions of less than 200 cm [79 in], DOE expects diffusion to dominate contaminate releases from saltstone. However, the NRC staff expects that advective flow in the fracture system could be significant. DOE used 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 analysis period. The saltstone matrix has not been demonstrated to respond quickly, and response times are expected to 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 (i.e., if 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 because 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 in the flow calculations may also affect release rates and transport in the vadose zone. PA calculations could account for higher density fluid releases by performing transient simulations with coupled density-dependent flow and salt transport, where the modeled salt consists of the suite of dissolved constituents. Reviewing model support in this area is therefore important because it determines the magnitude of advective flow and transport.

Multiple sources of evidence to support model assumptions about chemical and physical

degradation mechanisms associated with changes in the hydraulic conductivity and effective diffusivity of saltstone grout and the vault concrete could include additional process modeling, experiments, laboratory and field studies, analogs, expert elicitation, and extensive literature reviews. Additional numerical modeling results to the exclusion of all other lines of evidence would not be considered as model support. In reviewing the results from the experiments or field studies, NRC staff should consider whether or not appropriate approaches were used and how well the results can be extrapolated to a bigger scale and/or over a longer time period. The technical review activity would be considered closed when (i) sufficient model support is available to limit the uncertainty in the predicted extent of chemical and physical degradation mechanisms associated with changes in the hydraulic conductivity and effective diffusivity of saltstone grout and the vault concrete such that the predicted dose meets the performance objectives or (ii) DOE demonstrates that such degradation does not significantly affect the predicted dose to a hypothetical member of the public.

3.1.4 Factor 3—Model Support

The NRC staff concluded that model support for various model assumptions is important to assessing whether DOE's disposal actions will be compliant with the performance objectives of 10 CFR 61.41:

> "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 (NRC, 2005a)."

Model support is essential for deterministic modeling that is not clearly conservative because uncertainties are not represented in a deterministic analysis. Model support provides confidence that the numerical model results adequately represent the behavior of the actual system. Independent lines of evidence are an excellent form of model support and can take the form of experimental results, documented analogs, or expert elicitation that substantiates the model assumptions used.

The five review areas of model support are discussed separately in the following five sections.

<u>Moisture Flow Through Fractures</u>

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 certain amount of water could 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, so that releases would be due to radionuclides diffusing against the direction of water movement. In the PORFLOW simulations of release from the saltstone and vaults, DOE used moisture characteristic curve data to describe moisture flow under unsaturated conditions (WSRC, 2005b). However, anomalously high increases in gas flow during the laboratory test have shown that this testing cannot be used as the basis for moisture characteristic curves for the numerical modeling. DOE assessed

the impact of using the affected 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 basecase results (WSRC, 2005a). As discussed in the TER, NRC staff should evaluate measurements that replace the previous relative permeability measurements or verify that DOE used a bounding value of relative permeability in its PA.

As discussed in Section 3.1.2, the steady-state flow model DOE used to evaluate the flow of water through the saltstone wasteform may underestimate flow through fractures if the matrix does not equilibrate quickly and the wasteform is initially wetter than it is at equilibrium. Therefore, NRC staff should evaluate any available information about the rate of equilibration of the saltstone water content or the saltstone's initial water content to determine whether the information supports DOE's conclusion that the saltstone would maintain too much capillary suction to allow fracture flow. However, 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 flowed from unsaturated soils through joints in the vaults surrounding the high-level waste tanks (Lockie, 2002). It is postulated that flow occurred primarily because of rapid influx of snowmelt 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: (i) heterogeneity in material properties; (ii) temporal variations in saturation, especially resulting from rapid transport of infiltration through root holes to the lower drainage layer; (iii) infilling of fractures and joints with porous media; (iv) offset of the essentially impermeable vault/saltstone on either side of the fractures; (v) variability in the aperture of the joints and fractures; (vi) sensitivity to grid size in the simulations, especially at the interface of the fracture and porous media; and (vii) 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 consider site-specific behavior.

This review activity should be done in coordination with the technical review activities of Factors 1 and 2. The status of the technical review activity would be considered closed when sufficient model support is provided to support the assumed moisture flow rate through fractures as represented in the PA.

<u>Waste Oxidation and Technetium Release</u>

Saltstone oxidation is important to meeting the performance objectives in 10 CFR 61.41 because oxidation can lead to increased releases of technetium from the wasteform. The NRC TER stated that reasonable assurance that the performance objectives would be met depended on several assumptions, including the assumption that DOE would perform more realistic modeling of waste oxidation and that the results would demonstrate that technetium release rates will be acceptable (see Factor 1).

The dry grout formula described in DOE's final waste determination contains blast furnace slag that creates chemically reducing conditions to immobilize technetium. Retention of technetium in grout made with blast furnace slag is supported by field lysimeter experiments that 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. Oxidation is expected to occur at saltstone surfaces, including the surfaces of cracks, where the saltstone is exposed to water or, potentially, to air. In an alternative conceptual model, an oxidized layer at saltstone surfaces could significantly increase technetium release. In DOE's basecase model, the primary mechanism for release of radionuclides from saltstone is diffusion. Because diffusive transport is directly proportional to the concentration gradient in the fluid phase and fluid phase concentrations of technetium are many orders of magnitude higher under oxidizing conditions than they are under reducing conditions, releases from the saltstone are expected to be sensitive to the oxidized layer formation rate. 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 assuming 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, this implies that the oxidized rind may be thicker than DOE estimated.

A conceptual model for waste oxidation and technetium release should be provided and reviewed by NRC staff. Model support may include any experimental or other evidence that saltstone will maintain a reducing environment considering that degradation (e.g., chemical and physical) is likely to be represented as a shrinking-core type of process at exposed surfaces and that oxidation may be significant even if the bulk material does not degrade significantly. In addition, measurements of the depth of penetration of the oxidation front in laboratory samples exposed to oxygenated water or, if appropriate, water and air, could be provided and reviewed. NRC staff must analyze and substantiate extrapolations to the larger saltstone wasteform as well as the longer performance period. Field experiments may provide direct evidence of both the rate of oxidation expected for saltstone samples buried at the saltstone disposal facility and the resulting release of technetium, such as field-scale saltstone lysimeter tests with and without slag. Longer term field scale results help support the predicted technetium release rates. In addition to providing information on field-scale release rates, the slag lysimeters may contain valuable information about the saltstone oxidation rate. If the lysimeter made with slag was exhumed, it may be possible to measure the penetration depth of the oxidation front to support the rate of saltstone oxidation under field conditions.

This review activity should performed in coordination with the technical review activity of Factor 1. The status of the technical review activity would be considered closed when sufficient model support is provided to buttress the assumed waste oxidation and technetium release as represented in the PA.

Extent and Frequency of Fractures

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. Due to uncertainties, the expected performance of the saltstone and concrete vault barrier is partially based on certain assumptions about the behavior of specific parts of the system.

As previously noted, wasteform and vault cracking due to hydration, precipitation, and recrystallization, cracking can also occur from settling and/or seismic forces, external static loading, and freeze–thaw cycling. An adequate technical basis must be provided for each fracture-forming degradation mechanism that is excluded for the PA. For example, calculations for external static loading, wasteform structural integrity, and, as previously discussed, the formation of expansive matrix phases, must consider the void spaces within the wasteform originating from interior structures such as circumference drain pipes and pillars made out of carbon steel.

If NRC staff is able to substantiate any extrapolations to the larger saltstone wasteform as well as the longer performance period, the degradation model assumptions could be supported by laboratory and field studies. Assumptions used in the calculations of flow through fractured vaults and saltstone include the occurrence of vertical fractures with a certain fracture width and spacing. These assumptions, such as the probability of straight vertical fractures versus a branching fracture network as observed in ceramics and concrete, may be substantiated by such information. Branching cracks, along with microcracks that result from mechanical and chemical (e.g., sulfate attack) effects, could lead to higher radionuclide releases compared to vertical fractures. Characterization information from Vault 1 could possibly be used because it has experienced cracking. Information from Vault 1 and other sources on vertical versus branching fractures could be documented. Moisture content and potential flow could be analyzed. Measurements and observations could be used to support the important assumption of the cracks pinching out on the top or bottom of the wasteform, halting flow. Field analyses could possibly support the assumption that shrinkage cracks are filled by the succeeding grout pour so as to preclude fracture flow. If the basis for the assumed degree of cracking in saltstone is cracking observations of Vault 1 or other material, differences between the chemical and physical properties of the wasteform and the concrete used in Vault 1 or other material should be addressed. In addition, the ability to observe small cracks and the possible implications of the existence of cracks that are too small to be observed should be reviewed with respect to the hydraulic properties of saltstone as well as saltstone oxidation.

This review activity should be coordinated with the technical review activity of Factor 2. The status of the technical review activity would be considered closed when sufficient model support is provided to support the assumed extent and frequency of fractures as represented in the PA.

<u>Lower Drainage Layer Plugging Rate</u>

The demonstration that the salt waste disposal can meet the performance objective for protection of the public is dependent on the 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 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. One of the degradation processes that reduces the effectiveness of the infiltration barrier is colloidal clay migration, which has the potential to move colloidal clay from backfill into the pores of the underlying drainage layers, particularly the lower drainage layer. The NRC TER (NRC, 2005a) states that colloidal clay transport is expected to 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. Because the integrity of the lower drainage layer

affects the potential of fractures in the vault to transport water, uncertainty in drainage layer behavior is expected to be important to performance.

As described in TER Section 3.2.5.1, the expected performance of the infiltration barrier is partially based on certain assumptions about the performance of the lower drainage layer. 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. Independent lines of evidence are an excellent form of model support and can take the form of experimental results, documented analogs, or expert elicitation that substantiates the model assumptions used. Experiments or field studies that simulate important aspects of processes related to plugging drainage layers through colloidal clay migration could supply results that could be used as model support. An engineered cap having undergone some form of accelerated degradation or a barrier-like natural analog may exist with a layering system comparable to that proposed at the saltstone disposal facility from which colloidal clay migration information may be obtained. In general, additional numerical modeling results without any other line of evidence would not be considered sufficient model support. In reviewing the results from the experiments or field studies, the NRC staff should consider whether or not appropriate approaches were used and how well the results can be extrapolated to a bigger scale and/or over a longer time period. At the end of the technical review, staff should understand how the lower drainage layer is designed; how the lower, vertical, and bottom drainage layers work as a system to divert water away from the vault; and how to prevent plugging by clay. Staff should also have information on what types of maintenance, if any, are required to keep the infiltration barrier working as designed for 10,000 years. This review activity should be coordinated with the technical review activities of Factors 4 and 5, both of which deal with the implementation of erosion and infiltration control features according to technically reviewed designs.

The status of the technical review activity would be considered closed when sufficient model support is provided to support the assumed plugging rate of the lower drainage layer of the engineered cap as represented in DOE's PA. Because the cap design has not been finalized and the cap is not expected to be installed until the vaults are filled, cap monitoring may not begin for several years. It is probable that the compliance monitoring plan will need to be updated as the disposal site design develops. Erosion and infiltration control designs may be finalized years after disposal actions involving grouting have been completed. Ongoing research may provide a basis for new designs that were not considered in the original waste determination. For example, a new design may not rely on a lower drainage layer for good performance. A revised compliance monitoring plan would reflect changes in the facility design and direct staff to evaluate model support for the features that significantly affect barrier performance.

<u>Infiltration Barrier Long-Term Performance</u>

The demonstration that the salt waste disposal can meet the performance objective for protection of the general population depends on the 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 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. Substantiating that the design will effectively limit infiltration to the extent and

for the length of time assumed in the PA is best achieved through model support. The engineered surface barrier design described in DOE's final waste determination is designed to reduce water fluxes that contact the waste using a multiple barrier system, including (i) vegetative ground cover, which causes higher evapotranspiration; (ii) a closure cap designed to shed water from the vaults using redundant drainage layers; and (iii) favorable hydraulic characteristics of the vault and wasteform that limit both advective transport and molecular diffusion within water. 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: (i) erosion, which removes topsoil and upper backfill; (ii) pine forest root penetration, which punctures the upper geosynthetic clay liner (GCL); and (iii) colloidal clay migration, which moves colloidal clay from backfill into the pores of the underlying drainage layers.

Due to uncertainties, the expected performance of the infiltration barrier is partially based on certain assumptions about the behavior of specific parts of the system. Independent lines of evidence are an excellent form of model support and can take the form of experimental results, documented analogs, or expert elicitation that substantiates the model assumptions used. Experiments which simulate important aspects of infiltration control and the degradation processes of infiltration barriers could supply results which could be used as model support. Engineered analogs (i.e., previously constructed engineered surface barriers) have been observed and studied for their effectiveness as an infiltration barrier under many different settings and conditions (e.g., arid, temperate, and humid; variations of severe precipitation; variations of artificial and vegetative ground cover; and variations of construction material). Studies to date have shown that barriers relying on large evapotranspiration due to a well-thought-out vegetation ground cover have shown better performance. Although there are many types of hydrologic barrier caps, some cap components may be represented in natural analogs. Natural analogs can provide the basis for designing engineered surface barriers that mimic favorable natural settings and provide insights into future environmental conditions and the effects on barrier performance. Model support would include results and conclusions of studies that reduce the uncertainties regarding degradation processes that diminish the effectiveness of infiltration barriers. A technical basis should demonstrate how the lower, vertical, and bottom drainage layers work as a system and why plugging of the lower drainage layer and flooding of the vertical drainage layer will not happen or does not matter. Experiments or field studies on infiltration processes of similar materials with comparable vegetation in a climate that is expected at the site could provide information to be used as model support. For example, a literature review may obtain data on the penetration strength of pine trees under different settings and in different strata. Model support also might include the results of field studies on an engineered cap having undergone some form of accelerated degradation or a barrier like natural analog. Field studies of engineered systems or natural analogs with layers similar to those in the proposed saltstone cap would be particularly useful. For example, studies of clay migration into gravel layers may provide model support for assumptions made about drainage layer plugging.

NRC staff should review any projected water balance analysis associated with the infiltration control design. Components of the water budget would include precipitation, surface drainage and runoff, evapotranspiration, subsurface drainage away from the vaults, cap water storage, and infiltration to the vaults. Attention should be paid to the seasonal variation in precipitation and evapotranspiration (growing versus dormant period). Calculations from which the rates of each component were derived should be provided for the staff to review and to assess the uncertainties involved. Staff should understand the likelihood of preferred pathways emerging

for infiltration due to desiccation of the closure cap material and the significance of such a process. Analyses of expected subsidence rates of the vaults or surface barriers should be reviewed, as should the expected effect on performance by biotic intrusion, seismic activity, or other sources that cause irregularities or changes to the infiltration and drainage design. If the engineered surface barrier is designed as one closure cap covering all vaults together, NRC staff should verify that changes in the gradation at the surface and within the hydrologic barrier due to nonuniform subsidence rates between the areas covering the vaults and the areas in between the vaults will not affect performance. If other surface barriers of the type and scale proposed in the final design exist by the time of the cap construction, studies on their development and effectiveness would help provide additional confidence. At the end of the review, the reviewer should understand how the engineered surface barrier is designed, how it will be constructed, and how it is intended to function. The reviewer also should determine what maintenance, if any, is required to maintain barrier function as designed for 10,000 years.

The status of the technical review activity would be considered closed when sufficient model support is provided to demonstrate long-term effectiveness of the engineered surface barrier as an infiltration barrier as represented in the PA. Depending on the technology and knowledge at the time of cap construction, drainage, runoff, and infiltration rates of the final closure cap may be measured and monitored after closure to fully substantiate the model assumption concerning infiltration to the vaults.

This factor part asks DOE to substantiate the long-term performance of the infiltration barrier within the closure cap. This can most effectively be done if DOE builds model support to substantiate assumptions made with regard to (i) the plugging rate of the lower drainage layer of the engineered surface barrier (Factor 3, Part 4) and (ii) substantiating the infiltration control design as effectively limiting infiltration to the extent and for the length of time assumed in DOE's PA (Factor 5). Factor 4 directs the reviewer to verify that the erosion control features are built in accordance with the final design. The erosion control aspect of the engineered surface barrier is critical to preventing damage to the infiltration control system and effectively limiting infiltration to the vaults. The reviews of these model support parts should be coordinated with this technical review of the infiltration barrier long-term performance. The status of this technical review activity could not be considered closed if the technical review areas for Factors 3(4), 5, and 4 were still under review and their status open.

3.1.5 Factor 5—Infiltration Control Design and Performance

NRC staff concluded in its TER that the infiltration control design is an important factor to assessing whether DOE's disposal actions will be compliant with the performance objectives of 10 CFR 61.41:

> "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 (NRC, 2005a)."

The SDF closure concept described in DOE's final waste determination (DOE, 2006) includes a thick, multilayer engineered cap covered by vegetation (bamboo, later pine trees) and a surface

drainage system. The engineered surface barrier serves two primary purposes: to limit infiltration to the waste and to limit erosion. 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. Monitoring activities related to erosion are discussed in Section 4.1.2.

The closure cap design described in Action Item 3 (August 17, 2005) of Westinghouse Savannah River Company (2005a) includes three functional components separated by backfill: (i) an erosion barrier, (ii) an upper drainage unit intended to shed water far from the vault, and (iii) a lower drainage unit intended to route water around the top and sides of the vault into a buried drainage system. The erosion barrier is discussed in Section 4.1.2. The 0.305-m [1-ft]-thick upper drainage layer will be placed on top of the upper GCL to form a lateral drainage layer and to provide the necessary confining pressures to allow the GCL to hydrate appropriately. The upper drainage layer will be hydraulically connected to the overall facility drainage system to divert and transport as much infiltrating water as possible through the upper drainage layer to the facility drainage system and away from the underlying vaults. A vegetative cover will be established to promote runoff, minimize erosion, and promote evapotranspiration. If it is determined that bamboo is a climax species that prevents or greatly slows the intrusion of pine trees, it will be planted as the final vegetative cover. Pine trees are typically assumed to be the most deeply rooted, naturally occurring climax plant species at SRS (eventually assumed to replace the bamboo in the long term). The 0.61-m [2-ft]-thick lower drainage layer will be placed on top of the lower GCL to form a lateral drainage layer and to provide the necessary confining pressures to allow the GCL to hydrate appropriately. The lower drainage layer will be sloped at the same slope as the vault roof and lower GCL (i.e., minimum of 2-percent slope). The lower drainage layer shall consist of coarse sand with a minimum saturated hydraulic conductivity of 1×10^{-1} cm/sec [1×10^{5} ft/yr] that is free of any materials deleterious to either the underlying GCL or overlying geotextile. The coarse sand drainage layer will extend out from the vaults or be hydraulically connected to drainage layers on the sides and at the base of vaults to divert and transport as much infiltrating water as possible away from the underlying vaults.

An integrated drainage system will be designed and built to divert surface runoff from the closure caps and lateral drainage from the upper drainage layers away from the disposal site. As planned, the drainage system would consist of riprap-lined ditches that intercept the gravel layer of the moisture barrier. The riprap layer is intended to resist the velocities and shear forces associated with surface flows. Usually, riprap layers consist of a mass of well-graded rocks that vary in size. The size and weight of the stones that make up the layer are decisive. The layers will be designed in accordance with NUREG–1623 (Design of Erosion Protection for Long-Term Stabilization) (Johnson, 2002). The riprap-lined ditches, constructed between individual closure caps and around the perimeter of the SDF, will direct the water away from the vaults and the SDF as a whole. The ditches will discharge into sedimentation basins as necessary for sediment control. The riprap for the ditches has not been sized yet, because the SDF is currently in the initial phase of its 30-year operation period. Because the vault layout plan has not yet been completed, a detailed drainage system layout cannot yet be produced.

Although the final infiltration control design does not need to conform with the specific technical details provided previously, these details provide NRC staff a reference point when comparing the final infiltration control design to the design described in DOE's final waste determination. A final infiltration control design is years away; nevertheless, any future DOE final infiltration

control plan must divert a significant portion of water away from the wasteform to be an effective waste isolation barrier. 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. This factor directs reviewers to verify that the infiltration control design effectively limits infiltration to the extent and for the length of time assumed in DOE's PA. This can be done most effectively if DOE builds model support to substantiate assumptions about the rate that the lower drainage layer of the engineered surface barrier is expected to be plugged (model support part 4) and about the long-term performance of the engineered cap as an infiltration barrier (model support part 5). For example, if DOE measures drainage, runoff, or infiltration rates of the final closure cap, NRC staff could use the data to evaluate whether the measured values indicate the cap is performing as designed. Similarly, if DOE performs a pilot-scale test of its cap design and collected information about drainage rate plugging, NRC staff could evaluate whether the results of the studies support the rate of plugging of the lower drainage layer DOE used in its PA.

Both model support Parts 4 and 5 directly address uncertainties related to the implementation and performance of the final infiltration control design. Factor 4 directs reviewers to verify that the erosion control features are built in accordance with the final design. The erosional control aspect of the engineered surface barrier is critical to preventing damage to the infiltration control system and effective in limiting infiltration to the vaults. The reviews of these model support parts should be coordinated with the review of the final infiltration control design from Factor 5. This technical review activity should not be considered closed if the technical review areas for Factors 3 (Part 4 and 5) and 4 are still open. In addition, DOE has stated that SRS expects to retain an independent professional engineer to certify that the SDF closure system has been constructed in accordance with the approved closure plan and the final drawings, plan, and specifications at the time of closure. This action assures that the cover is built as designed in terms of hydraulic properties and provide additional confidence to close this compliance monitoring activity.

Onsite observations during the construction of the engineered surface barrier(s) and surface drainage system will be the most appropriate means to observe that many of the specific technical details of the infiltration final control design are constructed according to plan and to verify that an effective infiltration control system has been implemented and will perform as designed. Specifics of the final infiltration control plan and records and measurements made in the laboratory or in the field to support performance claims will be evaluated by technical reviewers. However, because the infiltration control design has not been finalized, detailed discussions on review items are not found in this version of the monitoring plan. When a final infiltration control design has been developed, the compliance monitoring plan will be revised to describe monitoring activities relevant to the final infiltration control design in more detail.

3.1.6 Factor 6—Feed Tank Sampling

To ensure that the inventory of radionuclides within the SDF will not lead to doses that exceed the performance objective for protecting the general population, DOE intends to use its PA results in establishing waste acceptance criteria for the SPF feed tank (i.e., Tank 50). DOE has developed a sampling plan to characterize the waste stream input to the feed tank and to verify

compliance with its established waste acceptance criteria (Ketusky, 2005). DOE's sampling plan entails the use of both direct sampling and process knowledge. As noted in the NRC TER (NRC, 2005a), the staff expressed concerns with the reliance on process knowledge in estimating the inventory of radionuclides sent to the SPF because of potentially large uncertainties associated with such estimates. The staff also noted that DOE's plan was unclear as to whether the frequency at which samples will be taken is consistent with the frequency at which the various waste batches will be sent to the feed tank.

In the TER (NRC, 2005a), the staff included the following factor that needed to be verified to assess whether DOE will be in compliance with the performance objectives of 10 CFR 61.41:

> "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)."

At the beginning of monitoring, NRC staff should ask DOE to provide a current version of their sampling plan for the SPF feed tank. The NRC staff should evaluate the sampling plan to confirm that the use of this plan will adequately characterize the waste that is sent to the SPF. NRC staff should also request a description of the quality assurance procedures associated with the sampling of waste, and the staff should evaluate whether these procedures are adequate.

As part of the sampling plan review, NRC staff should evaluate whether every waste stream sent to the SPF is adequately characterized. It is expected that the salt waste will be sent to the SPF in batches from the feed tank, Tank 50. The salt waste in each of these batches needs to be well characterized; the most straightforward method to do this is to obtain a representative sample of the waste from Tank 50. Another possible approach for characterizing the feed tank waste is to sample all of the inputs to this tank for each batch. By measuring the concentration in and volume of each waste stream added to the feed tank for a particular batch, it is theoretically possible to determine the overall concentration and inventory in the tank by mass balance. The use of such an approach would require supporting information that demonstrates that this approach adequately predicts the concentration and inventory in the feed tank. NRC staff should also determine whether there are any other waste streams transferred to the SPF other than those from Tank 50. If so, these waste streams will also need to be characterized. The sampling plan for the feed tank waste also needs to account for waste heterogeneity. Either a method for obtaining a representative waste sample from the feed tank waste should be developed, or enough samples should be taken to capture the variability in the waste.

NRC staff should also review the methods used to quantify the radionuclides in the feed tank waste samples. Ideally, the concentration of all radionuclides present in the waste would be measured. However, because there are many radionuclides in the salt waste, this may be difficult to do. If DOE does not plan to measure all radionuclides present in the salt waste, NRC staff should request information about how the radionuclides that are not measured will be quantified. The HRRs should be measured in all waste samples because these radionuclides were determined to be the most risk significant, though the determination of which radionuclides are HRRs was based on assumptions about the relative inventory of the

radionuclides.

NRC staff should request the feed tank sampling data on an annual basis for as long as salt waste is disposed in the SDF. This data should include the concentration and inventory of each radionuclide in each batch of waste transferred to the SPF as well as the volume of each batch of salt waste. The method used to quantify each radionuclide should be specified along with the detection limit and analytical method uncertainty associated with the method. NRC staff should evaluate uncertainty associated with the concentration and volume measurements. In addition to uncertainty in the analytical methods used to quantify the concentration of the radionuclides in the waste sample, there is also uncertainty associated with sampling waste that is potentially heterogeneous. The uncertainty associated with sampling is not as easily quantified as the analytical method uncertainty and is expected to be more significant. NRC staff should evaluate the measurement data provided by DOE to understand how much uncertainty is associated with sampling. Staff should determine whether the sampling data provide sufficient evidence that the salt waste in each batch is well mixed or that the sampling program adequately captures variability in the concentrations of radionuclides in a given batch of salt waste. If not, then the concentration data for the radionuclides, and consequently the inventory estimations, will be highly uncertain.

NRC staff should independently confirm the inventories DOE calculated for the waste that is disposed of in the SDF using the feed tank sampling data DOE collected. This review should be coordinated with the data review of the radioactive inventory (Section 3.1.1.1).

When monitoring begins, the technical review of the feed tank sampling will have a status of open, and this issue will be closed when the last of the salt waste has been transferred to the SDF.

3.1.7 Factor 7—Tank 48 Wasteform

The salt waste located in Tank 48 contains 19,000 kg [4,190 lbs] of the organic salts potassium and cesium tetraphenylborate (TPB). The TPB was added to the tank during an unsuccessful attempt at treating the salt waste using an in-tank precipitation process. TPB can break down into benzene and other organic compounds. The rate of this degradation increases with temperature and with the presence of catalysts. Because benzene has a high vapor pressure and evaporates within the normal tank operating temperature range, there is a concern about the formation of flammable vapors and potential explosion hazard. In the draft waste determination for salt waste disposal at SRS (DOE, 2005a), DOE states that because of the explosion hazard associated with the organic materials in this waste, DOE does not believe that they can process this waste to remove radionuclides. Instead, DOE planned to send the Tank 48 waste to the SPF feed tank without any processing. DOE has subsequently indicated that they are pursuing alternate options for the treatment or removal of the organic compounds in the Tank 48 waste.

In its TER (NRC, 2005a), NRC staff expressed concerns that the presence of large amounts of organic materials in Tank 48 waste could cause a wasteform made with Tank 48 material to have different properties than a wasteform made with other salt waste. Because no tests were performed on a wasteform made with Tank 48 waste or a surrogate waste containing comparable amounts of TPB, it is unknown how well a wasteform made with Tank 48 waste will perform or what its physical properties would be. The TER states:

> "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."

The technical review of this factor cannot be performed until DOE determines what treatment option they are going to use, if any, for the Tank 48 waste. However, even if the Tank 48 waste is treated to remove the TPB and other associated organic compounds, this factor will still be an open issue, and a technical review will still need to be performed because the treatment process will not remove 100 percent of the organic compounds, and it is likely that the chemical composition of the Tank 48 waste could still differ significantly from the salt waste originating in other tanks. Once DOE has selected an approach for treating and disposing of the Tank 48 waste, NRC staff should request a description of this approach. This description should include detailed information about the treatment process that DOE plans to use, if any, as well as a description of the expected effect on any selected treatment process on the chemical and radiological constituents in the Tank 48 waste.

To evaluate the potential effects of the organic material in the Tank 48 waste on the performance of the wasteform, it is important to know the chemical composition of the final Tank 48 waste stream (i.e., after treatment and/or mixing with other waste streams). NRC staff should ask DOE to provide any existing characterization results for this waste stream as well as any future plans for additional characterization. NRC staff should evaluate whether this characterization will provide adequate information about the composition of the waste stream and whether the analytical methods selected are appropriate for measuring the constituents of interest. The characterization should be done for the final waste stream that is going to be used to create the wasteform, not the Tank 48 waste itself. If any treatment processes are implemented for the Tank 48 waste, the chemical composition of the treated waste should be studied. Additionally, if the Tank 48 waste is mixed with another waste stream, the chemistry of the combined waste stream needs to be understood.

To assess the future performance of the wasteform made with the waste stream originating in Tank 48, the effects of the organic compounds in this waste stream on the physical integrity of wasteform and the ability of wasteform to retain radionuclides must be known. This information should be obtained through experiments to measure the properties of a wasteform created using the actual Tank 48 waste stream or using a nonradioactive surrogate for the Tank 48 waste stream. The TER states that at a minimum, these lab experiments should include tests for hydraulic conductivity and effective diffusivity. These parameters should be measured for both intact and degraded wasteforms. Other laboratory experiments may also provide useful information about the properties of the wasteform and its expected long-term performance. For example, it may be useful to study how the presence of the organic materials affects the chemical performance of the wasteform. The presence of the organic compounds could result in an increased amount of leaching of radionuclides from the wasteform if the organic compounds form chemical complexes with the radionuclides and make them more mobile. In addition, it is possible that the organic compounds could affect the sorption of the radionuclides in the grout. It may also be useful to study degradation rates of the wasteform to determine whether the presence of organic materials in the wasteform could result in a less stable

wasteform that degrades more quickly.

Once DOE has selected an approach for treating and disposing of the Tank 48 waste, NRC staff should request DOE's plans to perform experiments that measure the physical properties of the wasteform made using the waste stream that originates in Tank 48. NRC staff should review the proposed studies to evaluate whether they will provide the required data about wasteform performance. NRC staff should also evaluate the methodology proposed for these studies to determine whether the analytical method chosen is appropriate for measuring the parameter of interest. Staff should determine whether the solution used to create the wasteform in the experiments is appropriate and is either the actual Tank 48 waste stream that will be transferred to the SPF or is an appropriate surrogate solution. If a surrogate solution is used to create the wasteform instead of the actual waste stream, staff should evaluate whether the composition of the surrogate solution accurately reflects the composition of the Tank 48 waste stream.

Once the experiments have been completed, NRC staff should evaluate the results. Staff should determine whether any of the measured wasteform properties were substantially different than the parameter values assumed for the wasteform in the PA. If a substantial difference is observed, staff should evaluate whether it would lead to a decrease in wasteform performance and the potential effect of this decrease on compliance with the performance objectives of 10 CFR 61.41.

In addition to concerns about long-term behavior of a wasteform created with Tank 48 waste, there are also safety hazards associated with this wasteform because of the potential formation of benzene and other flammable vapors. NRC staff should evaluate the potential for formation of flammable vapors during the treatment and disposal of Tank 48 waste, especially during the curing of the wasteform. When the saltstone is curing, the heat of cement hydration will cause elevated temperatures in the wasteform, which could lead to increased rates of formation of benzene and other flammable vapors. To resolve this issue, NRC staff should ask DOE to provide data demonstrating that there will not be significant formation of flammable vapors from TPB present in the wasteform in the temperature range expected during curing.

The issue Tank 48 wasteform performance will be open at the onset of monitoring. This issue will be closed after the information documented in this section has been provided to NRC to demonstrate that a wasteform made with waste originating in Tank 48 will perform adequately, and the Tank 48 waste has been disposed of safely.

3.1.8 Factor 8—Removal Efficiencies

As described in Section 3.1.1.1, DOE intends to remove radionuclides from salt waste with a two-phase, three-part approach that includes (i) DDA, (ii) a combination of DDA and treatment with the ARP and MCU, and (iii) treatment with the SWPF. The removal of radionuclides from salt waste directly affects the ability of the disposal facility to meet the performance objectives because the performance of the facility depends on the final inventory of radionuclides in the SDF. In addition, the amount of radionuclide removal directly affects the determination of whether doses have been reduced to levels that are ALARA, which is part of the performance objective of 61.41. For these reasons, removal efficiencies were identified as a key factor for monitoring. In the NRC TER, staff noted:

"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 an important factor in the determination that 10 CFR 61.41 and 10 CFR 61.42 can be met."

As described in Section 3.1.1.2, after the data review on radionuclide removal efficiencies is complete, the staff should compare the removal efficiencies that are actually achieved to the projected removal efficiencies used to support the waste determination (Table 3). In information

Table 3. Estimated Removal Efficiencies of Highly Radioactive Radionuclides (in Percent)*

Radionuclide	DDA†	ARP/MCU‡§	SWPF‖	Overall
Cs-137/Ba-137m	50	91	>99	99^
Sr-90/Y-90	66	>99	>99	>99^
Se-79	40^	59^	59§	60^#
Tc-99	4.2^	6.3^	6.3§	5.7^
I-129	0.03^	0.05^	0.05§	<0.05^
Pu-238	63	98.1	95.5	94^
Pu-239	59	96.4	91.6	91^
Am-241	66	99.3	99.7	>99^
Am-242m	66	99	99.8	>99^
Cm-244	66	99.3	99.8	99^

*Based on Table 2 of the Technical Evaluation Report [U.S. Nuclear Regulatory Commission (NRC). "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." Washington, DC: NRC. November 2005].
†DDA = deliquification, dissolution, and adjustment
‡ARP/MCU = actinide removal process/modular caustic side solvent extraction
§For some radionuclides, there was insufficient information available to estimate the removal efficiency
‖SWPF = Salt waste processing facility
^Removal efficiency estimated by NRC staff. All other removal efficiencies were predicted by the U.S. Department of Energy [Westinghouse Savannah River Company (WSRC). "Radionuclides in SRS Salt Waste." CBU–PIT–2005–00195. Rev. 0. Aiken, South Carolina: WSRC. 2005].
#The overall removal efficiency appears to be greater than the removal efficiencies of individual processes

supporting the waste determination, DOE predicted treatment efficiencies for most of the HRRs identified in the waste determination (WSRC, 2005c). As explained in the TER, DOE did not provide predicted treatment efficiencies for all of the HRRs, and removal efficiencies for the remaining radionuclides were estimated by NRC staff. The process staff used to estimate the remaining removal efficiencies is explained in Section 3.6 of the TER.

The removal efficiencies used to support the waste determination were best estimates with associated uncertainties. DOE estimated that the overall removal efficiency of the DDA process, based on the deliquification and settling steps, may vary by approximately ±20 percent

of the nominal value for both soluble and insoluble radionuclides (WSRC, 2005c). 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 up to 25 percent for Pu-238 and Pu-239. Projected uncertainties in the efficiency of removal of radionuclides by the SWPF are less than 1 percent for Sr-90 and Cs-137 and approximately 5 percent or less for Pu-238, Pu-239, Am-241, and Cm-244.

Removal efficiencies are expected to vary from batch to batch because of differing waste characteristics. Because the removal efficiencies were not presented as upper bounds, failure to meet the projected removal efficiencies will not necessarily cause the SDF to fail to meet the performance objectives. An isolated low removal efficiency may not be significant unless the resulting concentration of an HRR in the saltstone waste is significantly higher than the projected concentration (Table 2). However, consistently low removal efficiencies or trends of declining removal efficiencies may be of more significance, especially if they differ from the projected removal efficiencies by more than the expected uncertainty. If the actual removal efficiency for one or more radionuclides is consistently lower than the projected removal efficiency, the staff should determine whether the lower removal efficiency is expected to have a significant effect on radionuclide inventory and the resulting dose to a hypothetical receptor. If so, the reviewer should ask DOE to demonstrate that the facility will be able to meet the performance objectives with the new projected inventory.

In addition to the reported uncertainties in each waste treatment processes, DOE also estimated that the concentration of sludge that becomes entrained in salt waste during the DDA dissolution step could vary from approximately 0 mg [0 oz] of sludge per liter salt waste to several grams of entrained sludge per liter salt waste (several tenths of an ounce of sludge per gallon salt waste) (WSRC, 2005b). When computing the expected inventory in the SDF, DOE based DDA effluent concentrations on an estimate that 600 mg of sludge would be entrained per liter of salt waste [0.08 oz/gal] (d'Entremont and Drumm, 2005). However, when estimating DDA removal efficiencies, DOE based the estimate on the deliquification and settling steps alone. Therefore, the large uncertainty in the amount of sludge that will be entrained in salt waste during salt waste dissolution and removal from the tank is not included in DOE's reported DDA radionuclide removal efficiencies. Unlike the ARP/MCU and HRR processes, which include a filtering step that is expected to remove essentially all of the sludge that is entrained in salt waste during removal of the salt waste from the tanks, the DDA process is expected to remove only about 66 percent of the entrained sludge from the salt waste. Therefore, the sludge entrained in waste treated only with the DDA process is expected to contribute a significant fraction of some of the relatively insoluble HRRs in the SDF inventory. For example, 77 percent of the Sr-90, 28 percent of the Pu-238, and 14 percent of the Am-241 in the final SDF inventory is expected to be contributed by the DDA waste (WSRC, 2005c). Thus the amount of sludge entrained during DDA processing could have significantly affect the final inventories of several HRRs.

As part of monitoring radionuclide removal efficiencies, NRC staff should request an estimate of the amount of sludge entrained in the salt waste during the DDA process. If significantly more than 600 mg [0.02 oz] of sludge is entrained per liter [0.26 gal] of salt waste, staff should determine whether the concentrations and inventories of sludge-associated radionuclides will be significantly greater than the concentrations and inventories used to support DOE's decision (tables 2 and 3 of this monitoring plan). This part of the review should be coordinated with the monitoring activities related to radionuclide inventories (Section 3.1.1.1). Highly-radioactive

radionuclides that are expected to be associated with sludge include Sr-90, the actinides, and in some cases, Se-79. In addition, if the amount of sludge entrained in salt waste during DDA processing is significantly greater than 600 mg/L [0.08 oz/gal], staff should coordinate with the review of radionuclide inventories (Section 3.1.1.1) and the review of compliance with 10 CFR 61.42 (Section 4.1.1) to determine whether the inventory of Sn-126 is expected to remain consistent with meeting the 10 CFR 61.42 performance objective (also see Section 3.2.18 of the TER).

In addition to the numerical dose limit, staff should consider whether failure to meet the predicted removal efficiencies compromises DOE's ability to demonstrate that the ALARA requirement of the 10 CFR 61.41 performance objective has been met. In general, it is not expected that staff should need to reevaluate compliance with the ALARA requirement unless removal efficiencies of several batches of waste are outside of the expected range of uncertainty. In that case, staff should ask DOE to explain whether removal efficiencies could be improved with application of an alternative process or improvements to the selected process. If improvements are not implemented, the staff should ask DOE to explain why the low achieved removal efficiencies are consistent with the ALARA requirement of 10 CFR 61.41.

The agency's activities with respect to monitoring DOE's radionuclide removal efficiencies are expected to continue as long as waste is transferred to the SDF. The monitoring activities discussed in this section relate to the treatment process that DOE discussed in the draft waste determination. If DOE changes its treatment processes, NRC staff will need to adapt these review areas to the new treatment processes. In addition, if the treatment plan is changed, the reviewer should request the information necessary to verify that the new treatment processes will exceed the radionuclide removal capabilities of the processes on which the draft waste determination was based. If the new processes do not exceed the radionuclide removal capabilities of the processes discussed in the draft waste determination, the reviewer should ask DOE to explain how the new treatment processes and resulting radionuclide inventory can consistently meet the numerical performance objectives and the ALARA requirement of 10 CFR 61.41.

3.1.9 Performance Assessment Process Review

The Manual for DOE Order 435.1 (DOE M 345.1-1) requires that its PA be reviewed and revised when there are changes in its wasteform or containers, radionuclide inventories, facility design or operation, closure concepts, or there is an improved understanding of facility performance. In addition, DOE M 435.1-1 indicates that, on at least an annual basis, DOE should determine the continued adequacy of its PA considering the results of data collected and analyses from research, field studies, and monitoring undertaken. Given the requirements of DOE Order 435.1, periodic revisions to DOE PA are expected to be made throughout its waste disposal activities. Given the nature and extent of the key factors identified in the NRC TER, it is likely that future revisions or updates to the DOE PA will include modifications to both parameters and conceptual models. Changes to the assumed land-use scenario are less likely, but are possible if needed to demonstrate that the performance objectives will be met. Changes to the conceptual model and/or parameter values may be especially warranted if there are changes in the engineered system.

DOE PA is a critical element of its demonstration of compliance with the performance objectives

for 10 CFR 61.41 and 42 because it forms the primary basis for concluding that the performance objective will be met. DOE completed an initial PA for the SDF in 1992. This PA contained an inventory based on the in-tank precipitation process being used as the waste treatment process. DOE Order 5820.2A provided the compliance criteria. The 1992 PA quantitatively evaluated impacts to members of the public assuming a drinking water exposure scenario. Impacts from air exposure and impacts to an inadvertent intruder were assessed qualitatively. PORFLOW (Analytical and Computational Research, Inc., 2002) was the principal computer code used in the 1992 PA. The 1992 PA was updated through a special analysis completed in 2002, which used an inventory based on a low-curie salt feed. Compliance was demonstrated in this analysis against the criteria in DOE Order 435.1 and 10 CFR Part 61. Similar to the 1992 PA, the 2002 Special Analysis evaluated impacts to a member of the public by assuming a drinking water exposure scenario. However, impacts from air exposure and impacts to an inadvertent intruder were quantitatively evaluated. PATHRAE (Shuman and Merrell, 1987) was the primary computer code used in the 2002 Special Analysis.

As part of its performance demonstration for the salt waste disposal, DOE again updated its PA with a 2005 Special Analysis, which was intended to update the 1992 PA and supersede the 2002 Special Analysis. The primary purpose of the 2005 Special Analysis was to update the inventory and evaluate SDF against criteria in DOE Order 435.1 and 10 CFR Part 61 considering an all-pathways exposure scenario. The 2005 Special Analysis consisted of linking three computer codes to evaluate impacts to a member of the public and separate quantitative analyses to evaluate impacts from exposure from the air pathway and impacts to an inadvertent intruder. In terms of evaluating impacts to a member of the public, DOE used the following computer codes: (i) the Hydrologic Evaluation of Landfill Performance (HELP) code (EPA, 1994a,b) was used to calculate infiltration into the upper cover layers (through the upper geosynthetic liner); (ii) the PORFLOW code (Analytical and Computational Research, Inc., 2002) was used to calculate groundwater transport through the lower cover layers (i.e., below the upper geosynthetic liner), the SDF, the unsaturated zone, and the saturated zone to a hypothetical well 100 m [330 ft] from the SDF; and (iii) the LADTAP XL© spreadsheet code (Simpkins, 2004) was used to calculate doses to a hypothetical residential farmer. The air pathways exposure analysis was performed separately using a sequence of two computer codes: (i) the PORFLOW code was used to calculate gaseous diffusion from the vault and (ii) the CAP88 code (EPA, 2002) was then used to calculate air transport and resulting doses.

Because the PA is important to the performance demonstration, NRC staff should evaluate subsequent revisions and updates to it. Based on DOE’s prior efforts at updating the PA for the SDF, it appears that updates or revisions may occur as either a special analysis or a revision to the PA. In either case, NRC staff should evaluate updates in terms of their effects on the performance demonstration. Presumably revisions to the PA should allow greater confidence in its results; however, it is conceivable that revisions could create new or even greater uncertainties. Therefore, NRC staff will need to determine whether the revisions to the PA enhance confidence in the performance demonstration.

To evaluate revisions made to the PA model(s), the staff should use an approach similar to that used in the original technical evaluation of the waste disposal actions. NRC staff should use its own independent assessment to the extent practical. Sensitivity and uncertainty analyses can be especially useful in determining the effects and importance of changes to the PA model(s). To date, DOE PAs have been deterministic. If future revisions or updates to the PA rely on deterministic analyses, the staff will need to ensure that results provide a demonstrably

conservative estimate of potential doses, or DOE has provided appropriate support and basis for key assumptions or parameters within the PA. Determining the important features of a deterministic analysis may be difficult, which is one reason why it may be advantageous for the staff to undertake its own independent assessment. In the future, DOE may place greater reliance on the use of probabilistic analyses. If so, staff will need to ensure that appropriate parameter distributions are used and that the analyses appropriately account for correlation between parameters. There are a number of available guidance documents about conducting probabilistic dose analyses (NRC, 2006, 2000). Because of the interrelationship between models and parameters within the PA, any changes to individual parameters and models may require an evaluation of the whole PA. In addition, the NRC staff should verify that issues identified in DOE's basecase analysis TER are addressed. Specifically, as discussed in the TER, a revised analysis should (i) be based on the projected average vault inventory; (ii) account for plume overlap from multiple disposal vaults; (iii) account for the expected magnitude and timing of climate change from the natural cycling of climates; (iv) account for the expected magnitude and rate of oxidation of saltstone; (v) account for liquid and gas flow in fractures that are expected to develop during the performance period; and (vi) use bounding values for relative permeability of concrete and saltstone or be based on new moisture characteristic curve measurements for concrete and saltstone rather than the relative permeability measurements used to support the 2005 Special Analysis (Cook et al., 2005).

To facilitate its review, the staff should request that DOE clearly identify changes made to the PA model(s), the reason for the change, and the effects of the change. Previous DOE updates to its PA have not always included a clear and traceable documentation of the changes from one version to the next. Accordingly, a table with side-by-side comparison of old and new parameters or models would be helpful. DOE should be also requested to describe any efforts to verify model results if new computer codes are being introduced into the PA (e.g., benchmarking). Further, for any codes that DOE has developed, DOE should be requested to describe its computer development process, including quality assurance procedures.

It is anticipated that DOE will continue to revise its PA throughout its waste disposal activities and into the facility closure. DOE will likely develop a final PA to represent site conditions and facility construction with the final cover and inventory. Thus, NRC monitoring activities related to reviewing DOE's PA are expected to continue through closure of the SDF.

3.2 Onsite Observations

3.2.1 Experiments

As discussed in the TER for salt waste disposal at SRS (NRC, 2005a), DOE's PA included key assumptions in the following areas: (i) wasteform and vault degradation, (ii) the effectiveness of infiltration and erosion controls, and (iii) estimation of the radiological inventory. NRC staff concluded assumptions in these areas are important to assessing compliance with the performance objectives because sensitivity analyses performed by DOE demonstrated that alternate assumptions in each of these areas could lead to unacceptable results. Thus, validation of the assumptions made in each of these areas supports the demonstration that the performance objectives will be met. Assumptions about the radiological inventory are expected to be addressed by sampling of waste as it is sent to the SDF (Section 3.1.6 and 3.2.2). Validation of assumptions in the other two areas, wasteform and vault degradation and the

effectiveness of infiltration and erosion controls, may be based on field and laboratory experiments.

In the TER, the NRC staff noted several specific areas that would benefit from additional model support and made several recommendations for experiments that could be used to develop the necessary model support (NRC, 2005a). DOE is not required to perform any of the recommended experiments. In general, DOE may select the method of demonstrating compliance with the performance objectives, and additional experiments may not be needed to develop model support if subsequent versions of the PA for salt waste disposal do not depend on assumptions in the identified areas. However, it is anticipated that DOE may choose to perform experiments to provide model support in several key areas, including moisture flow in fractures, saltstone oxidation, the extent and frequency of fractures with time, plugging of the lower drainage layer, and long-term cap performance.

Because the flow of water through the saltstone wasteform is expected to be a key factor in determining whether the SDF will meet the 10 CFR 61.41 performance objective, NRC staff indicated several areas related to saltstone hydraulic properties that would benefit from additional support (NRC, 2005a). The NRC staff suggested that the uncertainty in the hydraulic conductivity of intact saltstone could be limited by laboratory experiments that included samples with a greater range of curing times than had been used in experiments that supported the waste determination. Similarly, because the leaching tests performed represented a relatively narrow range of grout compositions, additional laboratory tests may be needed to provide information about saltstone hydraulic conductivity and effective diffusivity if the grout formula is changed. In addition, because of the unknown effects of the organic chemicals in Tank 48 waste on the physical properties of saltstone, staff indicates in Factor 7 in the TER “future tests of the physical properties of samples that contain organic material similar to Tank 8 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.” In addition, in Factor 7 NRC recommended that laboratory experiments be performed to measure the hydraulic conductivity and effective diffusivity of saltstone made from Tank 48 waste.

In addition to supporting assumptions about hydraulic conductivity and effective diffusivity, laboratory tests may be needed to support a new moisture characteristic curve for saltstone. Because sensitivity analyses showed that alternate assumptions could lead to a fourfold increase in dose, the staff indicated that a revised basecase should either be based on more reliable moisture characteristic curve data or on the assumption that the relative permeability of the waste was equal to one independent of waste saturation.

Both laboratory and field-scale tests may be needed to provide information about the expected degradation of the saltstone and engineered barriers. For example, laboratory experiments in which saltstone is exposed to concentrations of oxygen that are greater than the concentration expected in the environment may be able to support models of long-term saltstone oxidation. In addition, experiments designed to simulate accelerated samples aging may be able to support assumptions about long-term vault and wasteform degradation. In many cases, however, field-scale studies may be needed to provide information about the as-emplaced characteristics of the saltstone and engineered barriers. For example, it is anticipated that field-scale tests will be needed to verify that the initial hydraulic properties of saltstone after placement in the SDF are similar to those measured in the laboratory. Similarly, field-scale tests may be needed to

support modeled closure cap performance and the potential plugging of the lower drainage layer with time, which was identified in the TER as a key factor to monitor to assess compliance with the performance objectives.

Onsite observations may include observing data-gathering activities, interviewing workers, and demonstrations by workers performing tasks, among other activities. Onsite observations of experiments are expected to focus on observations of field-scale experimental facilities, field measurements, sample collection, and interviews with workers who have performed field or laboratory experiments. Other activities, such as observation of laboratory experiments, may be performed in some instances, but are not expected to be a primary monitoring activity.

In general, the goal of each of the onsite observation activities related to experiments is to determine whether the results of the experiment being performed will apply to the technical area that would benefit from additional model support. For example, one of these areas is long-term degradation of the vaults and saltstone wasteform. Model support could be provided by creating sacrificial samples of vault and saltstone materials, burying them at the site, and retrieving them years later to measure the development of cracks, growth of expansive phases, thickness of the oxidized layer, moisture retention, hydraulic conductivity, and effective diffusivity. To provide results that apply to a model of SDF performance, the samples would need to be exposed to conditions similar to the as-emplaced conditions of saltstone at the SDF. Samples would need to be cured over a sufficient range in curing times and temperatures to include the conditions that could be experienced by the full-scale vault or saltstone waste. In addition, the experiment would need to account for differences between the samples and the saltstone wasteform, such as differences in the surface area-to-volume ratio. By observing the experimental facilities and interviewing the workers performing the experiments related to these and other topics, NRC staff should determine whether any conclusions based on the experiments would support modeling assumptions about the degradation of SDF vaults and the saltstone wasteform. In general, technical staff with expertise in the subject area should determine what aspects of the experiment are likely to be most important to establishing whether the results of a particular experiment are applicable to a model of SDF performance.

The experiments discussed in this section represent examples of experiments that could be done to provide model support for key areas of the salt waste PA analysis. As previously discussed, DOE is not required to perform any of the experiments discussed in this section. Similarly, DOE may choose to perform experiments in areas that were not discussed in this section. This discussion is not an exhaustive list of the types of experiments that may be used to address the technical areas identified in the list of monitoring factors in the TER.

Onsite observations of DOE experiments are expected to continue until adequate model support is provided for the PA analysis. Because Factor 3 addresses the technical areas requiring model support, onsite observations related to experiments should be coordinated with the review of Factor 3 (Section 3.1.7). As DOE revises the PA for the SDF, additional areas requiring model support may be identified. In general, the NRC staff may perform onsite observations of any experiments that are used to support the demonstration that the SDF will meet the 10 CFR Part 61, Subpart C, performance objectives. Monitoring activities related to DOE experiments are expected to continue until additional model support is no longer needed, which may occur before or after SDF closure. Several technical areas that may be addressed with laboratory tests, such as information about the hydraulic properties of saltstone samples made with a variety of curing times or information characterizing new moisture characteristic

curves, may be resolved before SDF closure. However, areas related to features that will not be built until shortly before closure, such as the erosion barrier or infiltration control cap, or for areas related to the long-term characteristics of engineered barriers, such as the long-term degradation of the vaults or the saltstone wasteform, may remain open after SDF closure.

3.2.2 Waste Sampling

As noted in Section 3.1.1.1, DOE used a sum-of-fractions approach to demonstrate compliance with 10 CFR 61.41; as a result, the SDF inventory directly impacts DOE's demonstration that the performance objectives of 10 CFR 61.41 can be met. To accurately determine the inventory that is disposed of in the SDF, the waste streams that are sent to SDF must be adequately sampled and characterized through a waste sampling program. The technical review of this program that will be performed as part of the monitoring process is described in Section 3.1.6. In addition to the technical review, NRC staff should perform an onsite observation of the waste sampling program to verify that the samples are being collected properly and that the data obtained from the analysis of these samples are valid.

As described in Section 3.1.6, in its final waste determination, DOE indicated it plans to use Tank 50 in the H-tank farm as the feed tank for the SPF. It is expected that waste will be transferred in batches from Tank 50 to the SPF. To accurately track the inventory of radionuclides that is transferred to the SPF, every batch of waste transferred to SPF from Tank 50 must be characterized. Prior to the first onsite observation, the NRC staff responsible for the waste sampling review should coordinate with the staff performing the technical review of Factor 6—Feed Tank Sampling (Section 3.1.6) to determine what waste sampling approach DOE plans to use and whether this approach is adequate.

The onsite observations of the waste sampling program should be done periodically while salt waste disposal operations are being carried out. Staff should review the methods used by DOE to determine both the concentration of radionuclides in the waste as well as the volumes of the waste streams because both the concentration and volume must be known to determine the inventory in the SDF. During the onsite observation, NRC staff should observe the sampling of waste to verify that the sampling methods and quality assurance plans are being followed. When observing the sampling of waste, NRC staff should also evaluate whether the method used to collect samples is likely to collect a well mixed and representative sample. The NRC staff conducting the onsite observation should also confirm that the location(s) and number of samples DOE collected are consistent with the established sampling plan, and NRC staff should confirm that all waste streams sent to the SPF are included in this sampling plan.

At the onset of monitoring, the onsite observations of the waste sampling program for the salt waste will be categorized as an open issue. This issue can be closed after waste disposal operations at SDF have ceased if NRC staff is satisfied that all of the waste streams disposed of in the SDF were adequately characterized.

3.2.3 Vault Construction

The concrete vault is expected to provide secondary containment of the contaminants, with primary containment expected to be provided by the saltstone wasteform. DOE analyses indicate that the performance of the concrete vault could be important in meeting the

performance objectives of 10 CFR 61.41. Thus it is necessary to understand the potential mechanisms for loss of integrity of the vault and to monitor the vault performance during onsite observations.

In developing the conceptual model for its PA, DOE assumed a specific facility design. Two vaults already have been built at the SDF. Vault 1 contains waste from previous waste disposal operations. Vault 4 was specifically constructed to receive waste from the salt waste process, but currently does not contain any waste. Documents supporting DOE’s waste determination indicate that, in addition to Vaults 1 and 4, DOE plans to construct up to 13 additional vaults made of reinforced concrete. Each of these new vaults is planned to be approximately 180 m [591 ft] long by 60 m [197 ft] wide by 7.6 m [25 ft] high. Each vault would be subdivided into 12 cells approximately 30 m [98 ft] by 30 m [98 ft]. DOE’s final waste determination indicates DOE plans to fill each cell to a height of approximately 7.5 m [25 ft] with saltstone and then pour a layer of clean grout on the saltstone to fill in the space between the saltstone and the vault roof.

During onsite observations, NRC staff will identify noticeable deviations from the vault design (e.g., in the configuration or in the location of the vaults), focusing on changes that could affect potential pathways for water into the vaults, such as penetrations through the top of the vault or joints through which water could infiltrate. In addition, NRC staff will note any observed changes to the design that could increase degradation of the vaults or the saltstone wasteform. For example, NRC staff should determine whether any reinforcing steel or metal piping has been added inside the vaults, because corrosion of any steel penetrations through the saltstone could cause the saltstone wasteform to crack. Similarly, placement of the vaults also may influence degradation. DOE’s final waste determination indicates the bottom of the saltstone vaults will be at least 1.5 m [5 ft] above the historical high water table beneath Z-Area. NRC staff should determine whether the vaults are at the designed elevation because innundation of the vaults by the water table could increase vault degradation and radionuclide release.

In its PA, DOE assumed the vaults would remain intact for at least 100 years, followed by a gradual degradation that increases the hydraulic conductivity of the vaults with time. As discussed in the TER, a number of chemical and physical processes could degrade a concrete vault and increase its hydraulic conductivity. The DOE PA assumes there would be no unmitigated degradation of the vaults during operation. NRC staff inspection of the vaults is planned as part of the NRC onsite observations to detect possible early vault deterioration, which could suggest a vault failure time earlier than was assumed in the DOE PA calculations. In the TER, the NRC staff indicated that corrosion of rebar and fill pipes within the vaults could contribute to the degradation of the vaults. Thus, vault inspection could focus on the locations of these materials, which presumably could be obtained from DOE vault design drawings. NRC staff will assess how any observed early vault deterioration or changes in design have been considered in DOE PA updates.

NRC staff monitoring activities initially will be limited to Vault 4 but will include the new vaults as they are constructed in the future. The NRC monitoring activities related to vault construction are expected to continue throughout the waste disposal period until the vaults are covered.

3.2.4 Grout Formulation and Placement

As discussed in Section 3.1.3, saltstone composition is expected to affect its hydraulic properties, including its hydraulic conductivity and effective diffusivity. In addition, the fraction

of cement used in the saltstone formulation is expected to significantly affect radionuclide sorption by affecting the long-term pH of saltstone pore water as well as the density and type of binding sites available for radionuclide sorption. Because saltstone composition is important to wasteform performance, the TER identifies saltstone composition as a technical issue to be monitored. As discussed in the TER, the DOE recommended the following saltstone composition for a nominal salt solution blend: 47-percent salt solution (29-percent salt), 25-percent Class F fly ash, 3-percent Type II Portland cement or lime, and 25-percent Grade 120 slag (all percent values based on weight). If the final saltstone composition is significantly different from the nominal saltstone composition, additional tests will be needed to provide an adequate basis for the hydraulic conductivity and effective diffusivity of saltstone made with the new formulation (Factor 2). As part of the onsite observations, NRC staff should verify that the saltstone formulation used is consistent with the formulation described in the waste determination or with a new formulation reviewed by NRC staff as part of the monitoring activities described in Section 3.1.3.

In addition to the grout composition, curing conditions are also expected to significantly affect the as-emplaced conditions of saltstone (see Section 3.1.3) and its long-term performance. Because the samples measured for hydraulic conductivity to support the values used in the PA had a small range of curing times, the TER indicates that additional laboratory measurements should be made to provide a better basis for hydraulic conductivity values (Factor 2). Thus, staff performing onsite observations should coordinate with reviewers evaluating the basis for hydraulic conductivity values to verify that the curing conditions of the saltstone wasteform are not significantly different from the curing conditions of the laboratory samples that were measured to support hydraulic conductivity values. For example, NRC staff may ask knowledgeable DOE staff about measures that were taken to control temperature and humidity during curing.

Another important assumption in the DOE PA modeling is that the saltstone will have redox conditions sufficient to mitigate the release of Tc-99 due to the presence of ground blast furnace slag in the grout formulation. The hydration of slag in the grout mixture 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. Technetium is believed to react with the sulfide to form Tc_3S_{10} (Lukens, et al., 2005), mitigating the potential release of technetium through groundwater pathways.

The effectiveness of blast furnace slag in mitigating technetium release would depend not only on its relative amount in the grout formulation, but also on its sulfide content. The sulfide sulfur content of commercial blast furnace slag varies, typically from 0.7 to 1.1 percent by weight. The American Society of Testing and Materials (ASTM) standard C–989 (ASTM International, 2006) sets a maximum sulfide content limit of 2.5 percent by weight in Grade 120 slag, but specifies no minimum sulfide limit. Thus, DOE will need to specify in its vendor slag specifications a value for minimum sulfide concentration that it considers sufficient to ensure reducing conditions in the grout will occur. Because the minimum slag content required to achieve reducing conditions is uncertain, laboratory measurements may be needed to determine the reducing capacity of the grout and demonstrate that the minimum value selected for the specific DOE grout formulation will result in a reducing condition sufficient to mitigate the release of technetium. A procedure for such measurements is presented in Kaplan, et al. (2005).

In addition, DOE must show it implements a program of sampling, testing, and accepting

ground granulated blast furnace slag to ensure the slag conforms to DOE specifications and national standards, such as ASTM C–989. The verification program should incorporate a comprehensive record keeping system to include (i) plant operation records, (ii) ground granulated blast furnace slag production records, (iii) laboratory test results of composite or grab samples, and (iv) certification of shipping records. The sulfur sulfide content listed in the DOE vendor slag specification should be sufficient to ensure a reducing environment in the tank grout, as determined by laboratory measurements of the reducing capacity of the DOE grout formulation. The reviewer should ensure the reducing capacity of the grout is consistent with that assumed in the oxidation modeling used to support Factor 1 (Section 3.1.2).

During onsite observations, NRC staff should evaluate the adequacy of the verification program pertaining to its supply of blast furnace slag. The NRC staff's evaluation should be based, to the extent practicable, on direct observation of ongoing activities and interviews with key DOE personnel. The observation should evaluate certain aspects of the program, such as

- Representativeness of the samples collected
- Adequacy of the analytical equipment
- Calibration of the analytical equipment
- Adequacy of verification records

NRC staff also may acquire grab samples of the blast furnace slag for confirmatory tests to ascertain the material meets the applicable specifications.

Although blast furnace slag is stable under most conditions, deterioration of the quality and chemical reactivity of the slag can occur if it is not stored properly or exposed to moisture. To minimize the degradation in the quality and chemical reactivity of the slag, the material must be stored in weather-tight silos or bins to prevent contact with moisture. During onsite observations, NRC staff will need to inspect and evaluate the adequacy of the silos or bins for slag and cementitious materials storage.

3.2.5 Engineered Surface Barrier Construction

In DOE's "Response to Action Items From Public Meetings Between NRC AND DOE to Discuss RAI [Request for Additional Information] for the Savannah River Site," dated September 2005 (WSRC, 2005a), DOE states:

> "Final closure of the entire SDF will occur at the end of the 30-year operational period. Final closure will consist of site preparation and construction of an integrated closure system composed of one or more closure caps installed over all the vaults and a drainage system. Final closure is primarily intended to physically stabilize the site, minimize infiltration, and provide an intruder deterrent. Final closure will take into account the vault characteristics and location, disposition of non-disposal structures and utilities, site topography and hydrogeology, potential exposure scenarios, and lessons learned implementing other closure systems, including other SRS facilities and Uranium Mill Tailings sites. Since the SDF is currently in the initial phase of its 30-year operation period, contains only two existing vaults, and will require an unspecified number of additional vaults on a yet to be determined layout, the information provided

herein on the closure cap is appropriately a scoping level concept."

Because the construction of the engineered surface barrier is not expected to commence for approximately 30 years, NRC monitoring activities in the near term are expected to be minimal. More definitive plans for NRC monitoring activities will need to be developed at a later time, when more is known about the scope and nature of the construction of the barrier.

3.2.6 Environmental Sampling

As described in Section 3.1.1.3, the review of environmental data DOE collected is an important part of assessing compliance with 10 CFR 61.41 because this data can indicate whether there is a significant amount of leaching of radionuclides into the subsurface. In particular, the data obtained from the groundwater monitoring wells in the Z-Area are especially important for determining whether significant leaching is occurring. Because the NRC staff are relying on the results of the DOE data in their assessment, NRC staff must observe the groundwater sampling process DOE uses to verify that it is adequate and that the data obtained through this process are valid.

As part of the onsite observation process for assessing the environmental sampling program, NRC staff should observe both the installation of the groundwater monitoring wells and the collection of samples from these wells. There are currently four wells in the Z-Area: one upgradient of the disposal facility and three downgradient of Vault 4. DOE plans to install three additional wells near Vault 1, and it is anticipated that DOE will install more wells as new vaults are constructed. During the initial onsite observation, NRC staff should examine the existing wells to verify that they were installed properly. It may be helpful to review the well completion reports for these wells prior to conducting this observation. As new wells are constructed, NRC staff should observe the installation to confirm that they are drilled, screened, and developed properly and that the protective casing will adequately protect the well. Alternatively, staff may request copies of well completion reports from either DOE or SCDHEC. During the environmental sampling program review, NRC staff should also observe the collection of groundwater samples at the SDF to confirm that samples are obtained from the wells in accordance with the established sampling plans. Staff should verify that the well is properly purged before samples are collected and that samples are collected in accordance with the quality assurance program. While observing the sampling of groundwater from the monitoring wells, NRC staff should also confirm that the well is still installed properly, that the well has not silted up, and that the protective casing has not begun to leak. Staff should review DOE's sampling protocol prior to site visits in which staff intend to monitor DOE's sampling.

Observations of the environmental sampling will be an open issue at the onset of monitoring and will remain open indefinitely. Onsite observations will likely need to be performed more frequently at the beginning of monitoring and will be performed less frequently and in less detail once the NRC staff has verified the adequacy of the DOE environmental sampling program. For example, wells that have already been installed will likely only need to be observed once at the beginning of monitoring and will not need to be reobserved. However, NRC staff may continue to observe the installation of new wells in the Z-Area as more vaults are constructed at the SDF. It may also be necessary to perform additional observations in the event that increased levels of contaminants are observed in the groundwater samples. The results of the environmental sampling observations should be documented in the annual monitoring report

generated by NRC staff, and any areas where the environmental sampling program is not adequate should be noted.

3.2.7 Site Access Control

While performing onsite observations, NRC staff should assess the measures DOE takes to control access to the site. This assessment can be made through various means. One approach may be for the staff to interview key members of DOE's security department and review cognizant records and reports. In addition, staff may decide to tour portions of the facility to ensure that fences and other barriers designed to prevent access to the site are intact and performing as designed. Other approaches may be developed, over time, as NRC and DOE gain experience in this arena.

3.3 Potential Long-Term Compliance Monitoring

Probabilistic PAs may show results which have a low likelihood of occurring but relatively significant consequences. Epistemic uncertainties can include unexpected failure modes, design and construction errors, as well as analysis errors, which are difficult to bound in any manner. Significant low probability events may occur and cause changes to the overall conceptual model of the facility. A long-term environmental monitoring plan normally is designed and implemented to detect substantial deviations from expected disposal system performance after operations. Such a monitoring plan would be implemented after final facility closure, but would be developed before and at closure using and incorporating the results of data collected under the preclosure monitoring program. The environmental monitoring activities would focus on environmental sampling and data, usually monitoring the groundwater at various monitoring points. Potential long-term compliance monitoring activities may include changes in groundwater levels, in the groundwater flow direction, and hydrogeochemistry. If contaminants are in the saturated zone due to intentional or unintentional preoperational releases, the plume may be used to monitor some of the changes in the groundwater system. Environmental monitoring activities may also include tracking changes in the unsaturated zone system or changes in soil gases. Fauna and flora may serve as performance indicators. For example, deep-rooted plants could be sampled to determine any uptake of radionuclides.

Potential long-term compliance monitoring activities for the SDF may include more than environmental monitoring activities. The infiltration barrier could potentially be subject to long-term monitoring activities. If the related model assumptions pertaining to infiltration control can only be partially substantiated, then NRC staff may recommend longer term monitoring activities after the engineered surface barrier is built. Depending on the technology and knowledge at the time of cap construction, drainage, runoff, and infiltration rates of the final closure cap may be measured and monitored for a designated time period to fully substantiate the model assumption concerning infiltration to the vaults. For example, at present, test-pad lysimeters are generally reliable to test minimal infiltration rates of less than a few millimeters per year [few hundredths of an inch per year] for extended times.

4 MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.42—PROTECTION FROM INADVERTENT INTRUSION

10 CFR 61.42. Protection of individuals from inadvertent intrusion.

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

The primary focus of the U.S. Nuclear Regulatory Commission (NRC) monitoring activities, especially while waste disposal activities are ongoing will be to ensure that the U.S. Department of Energy (DOE) intruder analysis adequately demonstrates that the performance objective will be met. Many aspects of the monitoring activities carried out to assess compliance with the protection of the general population in Section 4 will be insightful for assessing compliance with protection from inadvertent intrusion. NRC plans to carry out three primary activities to assess compliance with 10 CFR 61.41: technical review (Section 4.1), onsite observations (Section 4.2), and long-term monitoring (Section 4.3).

4.1 Technical Reviews

4.1.1 Reliance on Affiliated Technical Review Activities

A factor important to assessing compliance with 10 CFR 61.42 is identified in the NRC's technical evaluation report (TER): Factor 4 on the erosion control design discussed in Section 4.1.2. The erosion control design is important to ensuring that 10 CFR 61.42 can be met, because the erosion controls affect the performance of barriers important to protecting an inadvertent intruder. For example, the erosion barrier layer has been designed to maintain a 3.2-m [10.5-ft] thickness of soil over the top of the vaults, which excludes the intruder resident farmer scenario if the erosion barrier performs as designed, because most agricultural or resident intruder scenarios consider a nominal excavation depth of 3 m [9.8 ft]. The thickness of the surface barrier prevents waste from being exhumed and prohibits ingestion pathways. In the intruder resident scenario, a home is assumed to be excavated above the disposal units and the receptors are exposed to direct radiation that is attenuated by the intact saltstone vault roof, the foundation of the house, and a remnant thickness of the closure cap after construction excavation. The magnitude of the intruder doses is strongly influenced by the amount of shielding provided by the vaults and closure cap.

Additional monitoring activities to assess compliance with 10 CFR 61.42 are similar to the technical review and onsite observation areas used to assess compliance with 10 CFR 61.41. Vault construction, grout formulation, waste sampling, and engineered surface barrier construction are being monitored through onsite observations while radioactive inventory, treatment efficiencies, and degradation of the wasteform and concrete vaults are being monitored through technical reviews to assess compliance with 10 CFR 61.41. Although the monitoring activities performed to assess compliance with 10 CFR 61.41 will provide information necessary to assess compliance with 10 CFR 61.42, the status of a monitoring activity (i.e., open, open-noncompliant, or closed) may not be the same for each performance objective. In most cases, monitoring activities that support compliance assessment for both 10

CFR 61.41 and 10 CFR 61.42 are expected to be closed for 61.42 when they are closed for 61.41; however, staff should evaluate whether there is any reason that information would have a different effect on the demonstration of compliance with 61.41 than it would on the demonstration of compliance with 61.42. For example, uncertainties in the Cs-137 inventory are expected to have a more significant effect on the predicted dose to a hypothetical intruder than on the predicted dose to a hypothetical member of the general population protected by 10 CFR 61.41. Similarly, as discussed in the TER and in Section 3.1.8 of this plan, significant deviations from the predicted inventory of Sn-126 are expected to have a more significant effect on the demonstration of compliance with 10 CFR 61.42 than they would on the demonstration of compliance with 10 CFR 61.41.

As with any compliance monitoring activity, the status could change from closed to open if new information becomes available or significant events that indicate the monitoring area should be reopened have occurred. For example, new insights into the performance of the engineered surface barrier or new upward estimates of the probable maximum precipitation (PMP) or on dramatic climate changes may indicate that the monitoring activity status would need to be changed. In addition, if minimal surveillance, monitoring, and custodial care are carried out after closure, NRC staff expects to be informed of changes to features in the immediate area that might affect the performance of the final erosion control design. These changes may include vegetation denudation at the surface due to fires or storms, erosional features caused by extreme precipitation events or long-term processes, or visible surface changes due to earthquakes or other geological processes.

4.1.2 Factor 4—Erosion Control Design

The final Saltstone Disposal Facility (SDF) closure concept includes a thick, multilayer engineered cap covered by vegetation (bamboo, later pine trees) and a surface drainage system. The drainage system will consist of riprap-lined ditches that intercept the gravel layer of the moisture barrier. Usually, riprap layers consist of a mass of well-graded rocks which vary in size. The riprap layer is intended to resist the velocities and shear forces associated with surface flows. The size and weight of the stones which make up the layer are decisive. These ditches will divert surface runoff and water intercepted by the moisture barrier away from the disposal site. The engineered surface barrier serves two primary purposes: to limit infiltration to the waste and to limit erosion. The erosion control barrier will be located approximately 1 m [3.3 ft] below the top of the final closure cap and will be placed on top of the middle backfill and overlying geotextile fabric to form a barrier to erosion and gully formation. In addition, the upper soil layer is designed to be stable during a calculated PMP event, such that a considerable margin of safety exists for erosion prevention. For the side slopes, erosion protection will be provided by a rock layer placed directly on top of the slopes.

The NRC staff concluded that the erosion control design is an important factor for assessing whether DOE's disposal actions will be compliant with the performance objectives of 10 CFR 61.42:

> "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" (NRC, 2005a).

In the preceding quote, the document (WSRC, 2005a) refers to "Response to Action Items from Public Meetings Between NRC and DOE to Discuss RAI [Request for Additional Information] for the Savannah River Site" by Westinghouse Savannah River Company in September 2005. Action Item 3 (August 17, 2005) presents technical and design details of the SDF closure cap and its erosion control design, including technical bases for the design details given (WSRC, 2005a). Although the design presented within this action item response is not intended to constitute the final design, it provides sufficient information for planning purposes and for evaluating the engineered surface barrier configurations relative to its constructability and functionality. NRC staff will compare future closure cap technical details and erosion control designs with these design details to ensure consistency with long-term erosion control design principles.

In 2002, NRC published NUREG–1623 (Johnson, 2002), which presents a series of methods, guidelines, and procedures that NRC staff consider to be acceptable for designing erosion protection at uranium mill tailings sites. Guidance is presented for the design of soil covers and slopes; design of rock riprap for slopes, channels, aprons, outlets, and stream banks; and for methods to determine sediment yield and acceptable construction specifications. The closure cap design described in Westinghouse Savannah River Company (WSRC) (2005a) closely follows NUREG–1623 when calculating the engineered surface barrier slope stability with respect to erosion and severe precipitation events, in addition to riprap design and placement. A technical review of the final closure cap design would need to provide technical bases that the designs are consistent with long-term erosion control design principles, and NUREG–1623 ("Design of Erosion Protection for Long-Term Stabilization") provides methods and procedures to support the technical bases.

The following paragraphs discuss significant areas of DOE's erosion control design as described in NRC (2005a). NRC staff found this design to provide adequate long-term stability. The top portion of the engineered surface barrier will be protected by a 1-m [3-ft] thick soil layer. DOE evaluated the top slope for erosional stability using the permissible velocity procedure discussed in NUREG–1623. As a conservative measure, a riprap layer under the top soil cover is part of the design to prevent further vertical erosion into the geotextile and drainage layers. Using the design procedure recommended in NUREG–1623, DOE computed the required rock size to resist further gullying. 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 rock size variations are not extreme. Examples of acceptable gradations are provided in NUREG–1623. DOE developed riprap gradations and layer thicknesses using criteria suggested in NUREG–1623.

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: (i) provide riprap of adequate size to be stable against overland (downslope) flows produced by the design storm, with allowances for turbulence along the downstream portion of the toe; (ii) 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; (iii) 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 (iv) provide an apron with adequate rock size to resist flows that will occur laterally along the apron.

To ensure 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 of Testing and Materials (ASTM). ASTM publishes and updates an Annual Book of ASTM Standards, and rock durability testing is usually performed using these standardized test methods. 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: (i) bulk specific gravity, (ii) absorption, (iii) sodium sulfate soundness, (iv) Los Angeles abrasion at 100 cycles, and (v) 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.

As described in NRC's TER, DOE indicated the erosion barrier consists of rocks of a certain size based upon the PMP and the methodology outlined by Abt and Johnson (1991) and Johnson (2002). Consistent with the recommendations of Johnson (2002) and ASTM International (1997), the rock shall be angular, shall have a minimum specific gravity of 2.65, and shall be considered durable if (i) the rock is dense, sound, resistant to abrasion, free of clays, and free of cracks, seams, and other defects as determined by a petrographic examination (ASTM International, 2003a) and (ii) specific gravity (ASTM International, 2004), absorption (ASTM International, 2004), sodium sulfate soundness (ASTM International, 2005), Los Angeles abrasion (ASTM International, 2003b), and Schmidt Rebound Hardness–ISRM Method (Johnson, 2002) tests are performed on the rock. Based upon these tests and the scoring methodology outlined by Johnson (2002), the rock shall have a quality score of 80 or greater. The uppermost soil layer of the closure cap shall consist of soils capable of supporting a vegetative cover (i.e., topsoil) and have a maximum 1.5-percent slope to provide stability that will prevent gully initiation based upon the methodology outlined by Johnson (2002). The toe riprap and the side slope riprap have been sized based upon the PMP and the methodology outlined by Johnson (2002).

The final erosion control design does not need to conform with the technical details provided previously; however, the status of the technical review activity would be stipulated as closed if the final erosion control design does not deviate significantly from the information submitted in the closure cap design details provided in WSRC (2005a). If, due to future design and construction realities, the engineered surface barrier design must change to provide site stability and the necessary erosion protection for a 10,000-year period, the erosion control portion of the design must be reexamined and reviewed to demonstrate consistency with long-term erosion control design principles as shown in the guidance provided in NUREG–1623 and similar documents referenced in this section. In addition, DOE has stated that SRS expects to retain an independent professional engineer to certify that the SDF closure system has been constructed in accordance with the approved closure plan and the final drawings, plan, and specifications at the time of closure. This action will provide additional confidence and support to close this compliance monitoring activity.

4.1.3 Performance Assessment Process Review

As discussed in Section 3.1.9, the DOE 1992 PA for the SDF only included a qualitative assessment of impacts to an inadvertent intruder. Subsequent updates to the performance assessment (PA), including the 2002 Special Analysis and the 2005 Special Analysis, have included a quantitative assessment. The 2005 Special Analysis, which forms the basis for demonstrating compliance with 10 CFR 61.42 in the draft waste determination, evaluated the impacts assuming a residential intruder scenario. DOE also evaluated an agricultural and drilling scenario as part of their sensitivity analysis to address NRC concerns regarding the long-term performance of the erosion control barriers. The analysis was conducted using an analytical tool specially developed by DOE that calculated radionuclide-specific concentrations that would be encountered by a residential intruder.

As discussed in the NRC TER, DOE appropriately demonstrated that the requirements for protecting inadvertent intruders will be achieved. However, future revisions or updates to the intruder analysis may be warranted if DOE changes its inventory and/or the SDF design. In its TER, the NRC noted that the performance of the erosion barrier could affect the performance demonstration because a significant reduction in the thickness of the cover prior to a significant reduction in the Cs-137 activity in the facility could exceed the dose requirements.

Because DOE has previously evaluated impacts to inadvertent intruders separately from its assessment of impacts to members of the public, future revisions or updates to the intruder analysis will also likely be a separate analysis. Thus, an update or revision to the intruder analysis may not accompany a revision or update to the PA (as discussed in Section 3.1.9). If there are no updates or revisions to the intruder analysis, the staff will need to assess whether or not it is needed. For example, as previously stated, changes to the inventory or facility design may warrant a reassessment of impacts to the inadvertent intruder. Further, changes to parameters or models within the PA may suggest a need for changes in the way that the intruder analyses are done. If there are updates to the intruder analysis, NRC staff should request that DOE describe the specific changes, the reasons for the changes, and the basis for any new assumptions made in the analysis.

As discussed in Section 3.1.9, in reviewing changes to the intruder analysis, the staff should carry out its own independent assessment to the extent practical. Staff should use available guidance, such as the NRC draft Standard Review Plan for Activities Related to U.S. Department of Energy Waste Determinations (NRC, 2006), to ensure a consistent and thorough evaluation of DOE’s analysis. As discussed in the NRC TER (NRC, 2005a), DOE used the projected inventory for Vault 4 in its sensitivity analysis. Even though the agricultural and driller scenarios are considered less likely than the residential scenario, the NRC assessment showed that by scaling up the inventory based on the average for the 14 vaults, the agricultural scenario would exceed the dose limit. This points to the need for ensuring that an appropriate inventory is used in the assessment. Use of the average inventory in this case would have been better than the projected inventory for Vault 4; however, in some cases, even use of the average may not be appropriate if there are areas of the facility (e.g., specific vaults) where higher activity of certain radionuclides (e.g., gamma emitters) will be disposed. Thus, in reviewing revisions to DOE’s inadvertent intruder analysis, NRC staff will need to ensure that an inventory appropriate for the given scenario has been used. Further, it is not appropriate for

DOE to account for the likelihood of the exposure occurring because this was accounted for in the regulations by having a higher dose limit.

Changes to the intruder analysis are expected to continue throughout DOE's waste disposal operations. DOE will likely develop a final intruder analysis to represent site conditions and facility construction with the final cover and inventory. Thus, NRC monitoring activities related to reviewing DOE's intruder analysis are expected to continue through closure of the SDF.

4.2 Onsite Observations

Waste sampling, grout formulation, site access control, engineered surface barrier construction, vault construction, observing experiments, and environmental sampling related to 10 CFR 61.41 evaluation are discussed in Section 3.2 and are pertinent to onsite observations that should be performed for evaluation of 10 CFR 61.42. If the status of technical reviews and onsite observation areas are closed under 10 CFR 61.41 and all are found to be compliant, then the status of the inadvertent intruder performance objective will be considered closed and compliant.

As with any compliance monitoring activity, the status could change from closed to open if new information becomes available or significant events occur that indicate that the monitoring area should be reopened. For example, if at some time in the future visible surface changes occur due to erosional processes, biotic intrusion, earthquakes, or other geological processes that could expose piping or other auxiliary equipment and potentially lead to increased doses or additional pathways of exposure, then NRC may consider reopening this technical review or onsite observation area to understand the effect of these changes on compliance with 10 CFR 61.42.

4.3 Potential Long-Term Compliance Monitoring

Probabilistic PAs may predict that some improbable events could cause significant doses to a hypothetical receptor. Epistemic uncertainties can include unexpected failure modes, design and construction errors, as well as analysis errors, which are difficult to bound in any manner. Hard-to-predict events may occur and cause changes to the overall conceptual model of the facility. A long-term environmental monitoring plan normally is designed and implemented to detect substantial deviations from expected disposal system performance after operations. Such a monitoring plan would be implemented after final facility closure, but be developed before and at closure using and incorporating the results of data collected under the preclosure monitoring program. The environmental monitoring activities would focus on environmental sampling and data, usually monitoring the groundwater at various monitoring points. Note that many of the monitoring activities for protection against the inadvertent intrusion will reflect parallel concerns in the protection of the general population (10 CFR 61.41) (see Section 3.3).

Potential long-term compliance monitoring activities for the SDF may include more than environmental monitoring activities. The erosion barrier could potentially be subject to long-term monitoring activities. If the related model assumptions pertaining to erosion control can only be partially substantiated, then NRC staff may recommend longer term monitoring activities after the engineered surface barrier is built. Depending on the technology and knowledge at the time of cap construction, erosion, subsidence, biotic intrusion, and/or a

changing vegetative ground surface of the final closure cap may be measured and monitored for a designated time period to fully substantiate the model assumption concerning long-term erosion barrier performance. These changes may include vegetation denudation at the surface due to fires or storms, erosional features caused by extreme precipitation events or long-term processes (e.g., gully formation or encroachment), or visible surface changes due to significant biotic intrusion, earthquakes, or other geological processes.

5 MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.43—PROTECTION DURING OPERATIONS

10 CFR 61.43. Protection of individuals during operations.

> "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."

5.1 Technical Reviews

10 CFR 61.43 is directly related to protection of individuals during operations. The U.S. Nuclear Regulatory Commission (NRC) interprets the term "operations" as those U.S. Department of Energy (DOE) activities related to waste removal, grouting, stabilization, observation, maintenance, or other similar activities. NRC intends to evaluate this key monitoring area from the time that DOE issues its final waste determination until the end of the institutional control period. Assuming that workers will most likely be performing duties on a controlled DOE site, under DOE's radiation protection program, the 50 mSv/yr [5 rem/yr] radiation worker dose limit will apply. For members of the public, including workers performing limited activities not covered under a DOE radiation protection program, the 1 mSv/yr [100 mrem/yr] dose limit for members of the public will apply. To evaluate this performance objective, NRC staff will review DOE's worker radiation records, the as low as reasonably achievable (ALARA) program, and offsite dose assessment methods and results to assess compliance with 10 CFR 61.43. NRC will use a graded approach to monitor DOE to ensure that workers and members of the public are protected.

5.1.1 Radiation Data

The performance objective in 10 CFR 61.43 for protection of individuals during operations requires that waste disposal activities be carried out such that the radiation protection standards set forth in 10 CFR Part 20 be met and that doses be maintained ALARA. DOE's approach to demonstrating protection of individuals during operations is to crosswalk the relevant DOE regulation or limit found in 10 CFR Part 835 and relevant DOE orders. NRC has agreed 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, NRC has agreed that DOE has applied a number of measures to ensure that exposure of individuals is maintained ALARA.

As with any compliance monitoring activity, the status of this monitoring area could change from closed to open if new information becomes available or significant events occur that indicate that the monitoring area should be reopened. For example, if significant new information is obtained that indicates a failure of radiation protection controls related to the disposal facility,

then NRC may consider reopening this technical review area to further investigate compliance with this performance objective.

NRC staff should review, on at least an annual basis, DOE reports and records that are related to worker and general public dose during waste disposal operations. NRC will request pertinent data, studies, reports, and other materials related to various waste disposal activities from DOE. As part of that request, NRC should request reports such as incident reports, annual site worker dose reports, and site environmental reports for review. NRC technical staff should review these reports to assess whether doses are less than the limits found in 10 CFR Part 20 and are ALARA.

5.1.2 Environmental Data

10 CFR 61.43 states that during operations, a facility needs to meet the standards set for radiation protection in 10 CFR Part 20, except for releases of radioactivity from the land disposal facility, which are governed by the 0.25 mSv [25 mrem] annual dose limit for a member of the public set in 10 CFR 61.41. 10 CFR Part 20 further specifies that the maximum annual dose that a member of the public can receive from airborne emissions is 0.10 mSv [10 mrem]. Compliance with most of the dose requirements for protection of individuals during operations is expected to be assessed through the use of dosimetry and the monitoring of radiation data and radiation records (Sections 5.1.1 and 5.2.1). However, the compliance with the dose limits from effluents must be demonstrated using measured concentrations in the effluents.

During operations associated with salt waste disposal at the SRS, the primary effluent of concern will be air emissions, because it is not expected that there will be significant releases to the subsurface from the waste in the saltstone vaults during the time of operations. Additionally, the release of radionuclides from the saltstone to the subsurface is being monitored in assessment of compliance with 10 CFR 61.41 (Sections 3.1.1.3 and 3.2.7). Any leaching of contaminants from the vaults observed while the SDF is still in operation may indicate that the ability of the wasteform to retain the radionuclides is worse than expected and that 10 CFR 61.41 may not be met.

DOE monitors the air quality at the SRS using air sampling stations located at the site boundary as well as in other locations throughout the site. In addition, DOE monitors the airborne effluents from operating facilities by sampling the emissions from the stacks. Because none of the air sampling stations are located near the SDF, the usefulness of the data obtained from the air sampling stations is limited. For example, if the data show an increase in the level of contamination present in the air at the site boundary, it would not be possible to determine whether the source of the contamination is from the salt waste disposal operation because of the presence of other radiological sources onsite. NRC staff review of environmental data to assess compliance with 10 CFR 61.43 should therefore be focused on the airborne effluent data from the operating facilities associated with salt waste processing and disposal.

At the onset of monitoring, NRC staff should request DOE’s air effluent monitoring plans for the facilities associated with the salt waste processing and disposal. These sampling plans should be evaluated to determine whether they will result in adequate characterization of the emissions from these facilities. In particular, NRC staff should evaluate DOE’s methodology to monitor the levels of radionuclides in the air emissions. For example, staff should determine whether

the stacks are monitored continuously or sampled periodically. If they are not monitored continuously, staff should evaluate whether the sampling frequency and timing is sufficient to capture any variability in the effluents. NRC staff should also assess whether the analytical methods selected will adequately measure the concentrations of all radionuclides of interest. Some of the radionuclides that are present in the salt waste, such as tritium, C-14, and I-129, are not expected to be associated with particles to a significant degree in the air phase, so it might not be appropriate to quantify them by using air filters. NRC staff should also request information about DOE's quality assurance program for the monitoring of air emissions from the operating facilities associated with the salt waste disposal and should examine the quality assurance plans to confirm that they are appropriate.

NRC staff should request air effluent data from the treatment and processing facilities for the salt waste from DOE annually during the operation of these facilities. The data requested should include the results of all measurements of the concentration or activity of radionuclides in the air effluents as well as the flow rates of the various effluent streams. In addition, staff should request information on the uncertainty associated with those measurements, the detection limits for the methods used, and the results of measurements taken as part of the quality assurance program (e.g., spike samples, duplicate samples). NRC staff should use this information to evaluate the quality of the data submitted. The air emission data should be tracked to determine whether there are any increases in the airborne emissions of radionuclides over time. The reason for any observed increases should be determined, and if possible, the process should be adjusted to reduce the level of emissions. NRC staff should also evaluate the air effluent data to determine whether the activity released in the air emissions could cause a member of the public located at the SRS site boundary to receive an annual dose of greater than 0.10 mSv [10 mrem] through the air pathway. One simple way to determine whether a particular effluent could cause an annual dose of greater than 0.10 mSv [10 mrem] to a member of the public located offsite is to conservatively assume that the member of the public will breathe air with concentrations of radionuclides equal to the concentration in the air effluent for the whole year. If the dose calculated during this simple analysis is not less than 0.10 mSv [10 mrem], then more complicated and realistic modeling of the atmospheric transport of the plumes from the salt waste processing and disposal facilities to the offsite receptor will have to be performed.

The review of environmental data to assess compliance with 10 CFR 61.43, the protection of individuals during operations, will be classified as an open issue at the onset of monitoring. When the operations associated with the salt waste processing and disposal cease, the issue can be closed. During operations, the air effluent data should be evaluated yearly to determine the potential dose to a member of the public, and the results of this evaluation should be included in the annual monitoring report. The evaluation of the sampling methodology and the quality assurance procedures will likely only need to be reviewed during the first year of monitoring unless these procedures are found to be inadequate or there are any substantial changes made to the procedures or to the facilities.

5.2 Onsite Observations

As discussed in Section 3.2.7, the NRC staff will review DOE's program to control access to the site. The effectiveness of access controls is pertinent to assessing compliance with the 10 CFR 61.43 performance objective with respect to determining the exposure pathways and

point of compliance for members of the public during operations. NRC staff also plans to perform onsite observations of DOE's radiation protection and environmental sampling programs as described in the following two sections.

5.2.1 Radiation Protection Program

To determine the validity of the various reports and records discussed in Section 5.1.1, NRC staff should review, with the aid of key DOE personnel, the Radiation Protection Program responsible for producing such reports and records while performing onsite observations.

Through discussions, interviews, and perhaps onsite tours with DOE or DOE contractor personnel, NRC staff should assess whether the programs and policies presented in the draft waste determination (DOE, 2005a) are indeed in effect during the operational period. Specifically, NRC staff should verify that personnel involved in the waste disposal operations are provided dosimetry and are familiar with the requirements of the Radiation Protection Program.

5.2.2 Environmental Sampling

As discussed in Section 5.1.2, inhalation is the primary potential pathway for exposure of a member of the public to radioactive effluent during salt waste operations, so NRC staff needs to monitor the data obtained by DOE's air effluent monitoring program to assess compliance with 10 CFR 61.43. As part of air emissions monitoring, NRC staff must perform onsite observations of air effluent sampling programs to verify that the measurements were obtained properly and that the data gathered through this sampling program is accurate and valid.

During the onsite observation, NRC staff should watch the collection of air effluent samples and verify that the sampling protocol is consistent with the sampling plan, that the Quality Assurance and Quality Confirmation procedures are being implemented according to the quality assurance plans, and that the air samplers are set up and calibrated properly, if necessary. NRC staff should also verify that all potential effluents are captured by DOE's air effluent monitoring program. For example, NRC staff should verify that this program includes all airborne effluents from all facilities associated with the salt waste processing and disposal. In addition, NRC staff should evaluate whether the sampling program adequately captures the variability in the effluent streams. For example, if a particular effluent stream is only sampled periodically, NRC staff should evaluate whether the concentrations of radionuclides in the waste stream are expected to be constant with time. If not, then it is possible that the effluent monitoring program is not adequately characterizing the potential effluents.

As was the case for the technical review of the environmental data (Section 5.1.2), this issue will be classified as open from the onset of monitoring until operations have ceased. While the SDF is operating, NRC staff should observe the air sampling program periodically. The initial observations are to be more frequent and more extensive, and later observations will be less frequent and less detailed once NRC staff determines that the environmental sampling program is adequate. However, NRC staff may need to perform detailed observations if there are any major configuration changes to the facilities or the air sampling equipment or protocol. The results of the onsite observations should be included in the annual monitoring reports, and any aspect of the air monitoring program found not to be adequate should be documented.

6 MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.44—SITE STABILITY

10 CFR 61.44. Stability of the disposal site after closure.

> “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.”

6.1 Technical Reviews

No factors important to assessing compliance specifically for 10 CFR 61.44 were identified in the U.S. Nuclear Regulatory Commission (NRC) technical evaluation report (TER) for the Savannah River Site salt waste disposal (NRC, 2005a); however, the key areas responsible for providing Saltstone Disposal Facility stability were identified as the structural integrity provided by the grout wasteforms and the concrete vaults and the erosion barrier layer designed within the engineered surface barrier. The wasteform and vaults will likely contain minimal void space; therefore, differential settlement and the associated negative effects on waste isolation would be eliminated, while the erosion protection design must be determined to be adequate to provide reasonable assurance of long-term stability of the closure cap for erosion control purposes. Monitoring activities to assess compliance with 10 CFR 61.44 will therefore be concentrated on these key features, which will be monitored within the technical review areas and onsite observation areas of other performance objectives. Vault construction, grout formulation, and engineered surface barrier construction will be monitored through onsite observations to assess compliance with 10 CFR 61.41, while the erosion control design will be monitored to assess compliance with 10 CFR 61.42. Compliance or noncompliance with the performance objective for 10 CFR 61.44 is associated with the status of the aforementioned monitoring activities. As with any compliance monitoring activity, the status could change from closed to open if new information becomes available or significant events have occurred that indicate the monitoring area should be reopened. For example, new insights into the structural performance of the grout or the predicted performance of erosion controls, or new upward estimates of the probable maximum precipitation (PMP) or the predicted effects of climate changes, may indicate that the monitoring activity status would need to be changed. In addition, if minimal surveillance, monitoring, and custodial care are carried out after closure, NRC staff expects to be informed of changes to features in the immediate area that might affect the stability of the disposal site. These changes may include vegetation denudation at the surface due to fires or storms; erosional features caused by extreme precipitation events or long-term processes; or visible surface changes due to significant biotic intrusion, earthquakes, or other geological processes.

6.2 Onsite Observations

In the TER, NRC concluded that DOE had appropriately demonstrated that the site stability requirement will be met based on the location of the SDF and the design. The SDF is located such that flooding or seismic impacts are not expected. Further, the saltstone grout wasteform and concrete vaults are expected to limit differential settlement. In addition, the engineered cap

is intended to protect the facility from the effects of erosion. However, the staff noted that erosion is the main disruptive process that could possibly influence stability of the facility.

To assess compliance with 10 CFR 61.44, NRC staff will visually observe the facility for obvious signs of degeneration of the facility. For example, evidence of ponded water on the cover surface may be a sign of differential settlement. Surface fractures may be evidence of underlying displacement. NRC staff should also consider lineaments, which may be best seen on aerial photos, for evidence of underlying displacement. Staff is encouraged to plan site visits to observe the facility after severe weather events (e.g., storms, tornados) to ascertain how well the facility is holding up. Staff should pay close attention to evidence of erosion (e.g., gully formation) forming on or around the facility. DOE is expected to carry out an active maintenance program for the facility through the end of the institutional control period; therefore, any obvious signs of facility degradation should be remediated. However, such degradation can provide insights into potential long-term facility performance. NRC staff also should discuss any remediation or maintenance activities with South Carolina Department of Health and Environmental Control.

This monitoring activity is not expected to begin in earnest until after all of the tanks have been grouted and the engineered cover placed over the facility. NRC staff monitoring activities in this area are expected to remain open indefinitely.

7 DOCUMENTATION OF FINDINGS AND IMPLEMENTATION PLAN REVISIONS

7.1 Onsite Observation Reports

The U.S. Nuclear Regulatory Commission (NRC) staff participating in onsite observations are expected to develop a short report after each site visit. This report (i) will provide a written summary of what was done and any findings from the visit; (ii) should include participant identification, a description of the activities undertaken, and any staff assessment; and (iii) should describe any issues that may warrant additional investigation through either the staff's technical review or future site visits.

The report should be narrative and describe the scope of the onsite observation effort, the activities observed, and any issues that may warrant further attention or follow-up. Activities that are deemed closed as a result of the onsite observation should be so designated in the report. The report of the onsite monitoring results will be provided to the U.S. Department of Energy (DOE) and the State of South Carolina by letter.

7.2 Annual Compliance Monitoring Reports

Staff findings for the various technical and onsite reviews completed throughout the year in an annual report.[3] It is anticipated that staff will develop a written report of its findings immediately following any onsite visits (see Section 7.1). In addition, the annual report will document the staff's technical reviews of data, its review of reports or analyses to address the key factors, and its review of any updates or revisions to DOE's performance assessment (PA).

In documenting its review, the staff will need to specifically describe what was reviewed, whether there is reasonable assurance that the performance objectives will be met (this will include a description of the specific performance objectives that apply), and the basis for the staff findings (e.g., independent analyses conducted by the staff, supporting studies, expert opinion). In addition, the staff should describe any recommended action (e.g., additional studies or analyses) that should be undertaken to close out open activities.

The annual monitoring report should also track whether monitoring activities discussed in Sections 2-6 of this report are considered closed, open, or open-noncompliant. In the report, the staff should (i) describe its basis for reopening any activity that was previously closed but is being reopened, and its expected plans for monitoring the activity; (ii) describe any new monitoring activities identified during the year and the basis for opening them; (iii) identify any open and open-noncompliant activities that are expected to be carried out for the upcoming year; and (iv) document any actions or results which might change the status of noncompliant activities. For activities that have closed, the staff should document its basis for closing the review and any conditions attached to a closed activity which would prompt reopening.

[3] It is anticipated that during the early phases of carrying out its monitoring activities, NRC will develop an annual report. However, as the monitoring program progresses and the number of monitoring activities diminish, the staff will need to reassess whether less frequent reporting is warranted.

It is envisioned that staff will meet with DOE and the State on an annual[4] basis to discuss the status of the monitoring program. This meeting could help determine which activities are still open, what actions are needed to close activities, and potential areas of concern. The staff may also want to use this meeting to identify potential revisions that need to be made to the monitoring program in the future.[5] Staff should describe conclusions and followup actions that come out of this meeting within the annual monitoring report.

Lastly, staff should describe any anticipated problems or problems that should be brought to the attention of management in the monitoring report. For example, the staff may have identified issues through either its technical review or onsite visits that are not significant enough to prevent the performance objectives from being met, but could affect the performance demonstration over time. As another example, the staff may want to tell management it has potential concerns about getting needed information to close out activities. The disposition of any issues raised during the previous year should be also described in the report.

Figure 1 shows the topical areas that are likely to be covered in the annual monitoring report. A copy of the report will be provided to DOE and the State for informational purposes. In addition, the report will be made publicly available on the NRC's website.

7.3 Noncompliance Letters

In accordance with the National Defense Authorization Act (NDAA), the NRC is required to inform DOE, the covered State, and Congress if it considers any of DOE's waste disposal actions to be not in compliance with the performance objectives of 10 CFR Part 61, Subpart C. The specific congressional committees that NRC is required to inform are the Committee on Armed Services, Energy and Commerce, and Appropriations in the House of Representatives and the Committee on Armed Services, Energy and Natural Resources, Environment and Public Works, and Appropriations in the Senate. NRC is required to make this notification as soon as practicable after discovery of noncompliant conditions. In addition, the noncompliance notification letter will be made publicly available on NRC's website.

As the staff carries out its monitoring activities laid out in this monitoring plan, it will determine whether DOE is in compliance with the performance objectives. As shown in Figure 2, there are two primary ways that DOE will be considered noncompliant: (i) if there are sufficient indications that the criteria for one or more performance objective is currently being exceeded or (ii) if there are sufficient indications that the criteria for one or more performance objectives could be exceeded in the future. Possible indications that the performance objectives are currently being exceeded would be environmental concentrations at locations where individuals could be exposed to a dose exceeding the dose criteria. Other possible indications that the performance objectives is currently being exceeded would be radiation doses to workers or members of the public that exceed the dose limit or evidence of structural failure of the disposal facility.

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[4]The frequency of these meetings is expected to change as implementation of the NRC monitoring program progresses. As the number of monitoring activities diminish, NRC staff will need to revisit whether less frequent meetings are warranted.

[5]This could also include determining the need for less frequent reporting and meetings.

Topical Areas Expected in
Annual Compliance Monitoring Report

Onsite Reviews
- Areas reviewed
- Findings
- Basis for findings
- Recommended actions

Technical Reviews
- Areas reviewed
- Findings
- Basis for findings
- Recommended actions

New and Reopened Activities
- Area of concern
- Significance to performance demonstration
- Expected monitoring activities

Open-Noncompliant Activities
- Basis for status
- Actions or results that might change the status

Summary of Annual Meeting

Revisions to the Monitoring Plan

Potential Problems

Issues Needing Management Attention

Figure 1. Topical Areas Expected in the Annual Monitoring Report

NRC cannot base noncompliance solely on contemporaneous noncompliance. First, given the nature of the highly engineered facilities involved, evidence of problems meeting the performance objectives may not be observable for hundreds of years in the future. Thus, solely relying upon observable system failure may not allow the NRC to make a timely notification as required by the NDAA. In addition, assessing compliance for some performance objectives (e.g., 10 CFR 61.42) is difficult to accomplish through direct observation. Thus, the second means by which NRC may make a finding of noncompliance is through predictive modeling that indicates one or more of the performance objectives may not be met in the future. An indication that DOE may exceed the performance objective(s) would be if key assumptions relied upon in its performance demonstration cannot be substantiated as previous technical information and evidence had indicated. “Key” in this sense means that without the assumption, the performance demonstration cannot be made. Another indication would be if trends in the data indicate that at some future time the performance objective criteria will be exceeded.

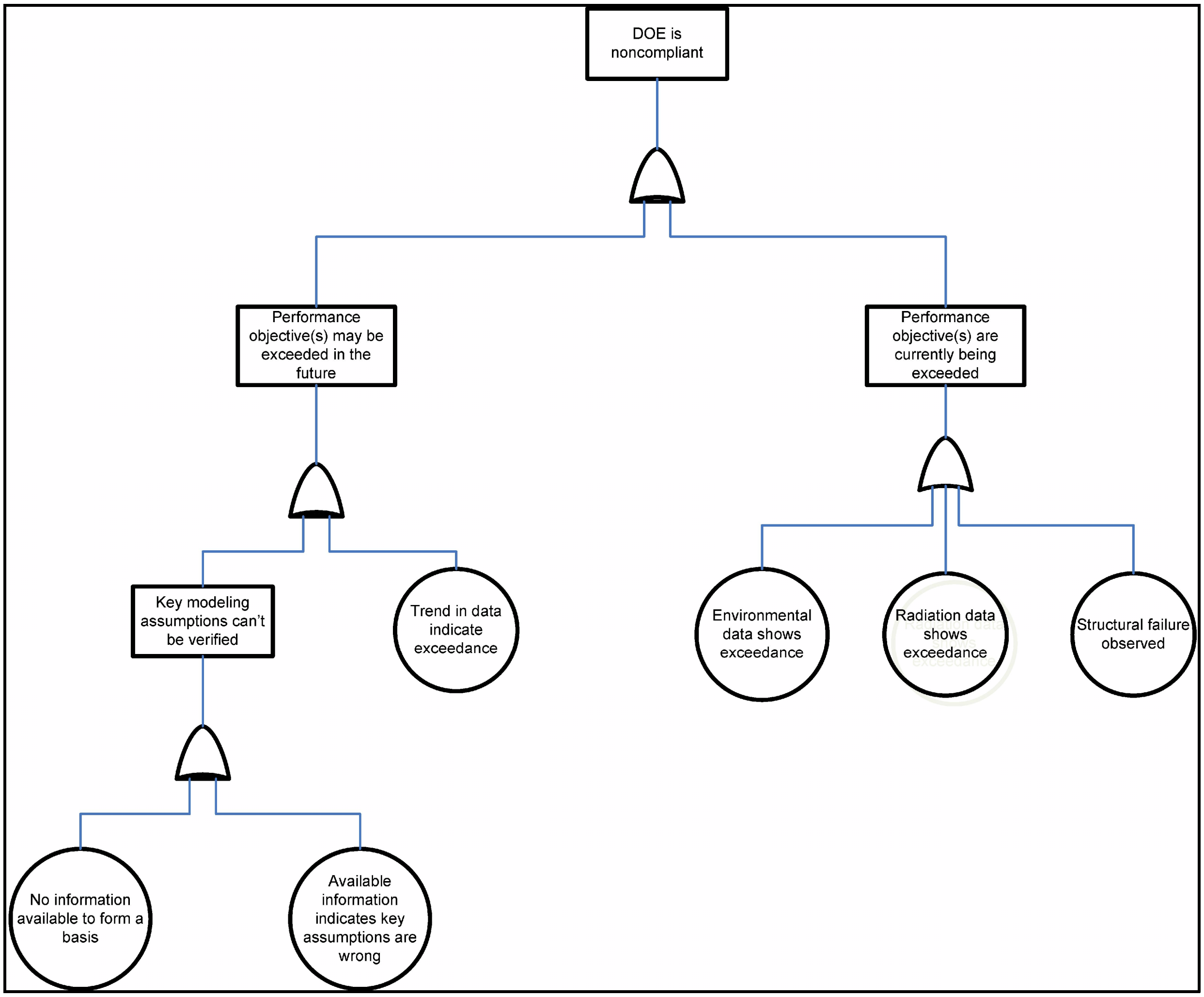

Figure 2. Potential Sources of Noncompliance by DOE

Given the different types of noncompliance, NRC anticipates using several different types of notification letters, listed in Table 4. A Type I letter would state that there are sufficient indications that DOE is currently not in compliance with one or more of the performance objectives. Within the letter, NRC would describe the performance objective(s)that DOE is not in compliance with and the basis for concluding that DOE is not in compliance. A Type II letter would state whether there is sufficient evidence to indicate that DOE will not be in compliance with the performance objective(s) at some point in the future. Again, this letter would identify the specific performance objective(s) that DOE is expected to exceed and the basis for the staff's conclusions. The third type of notification letter (Type III) would note whether there is an insufficient basis to conclude that DOE will be in compliance with one or more of the performance objectives. In this letter, NRC would identify the specific performance objectives in which a finding cannot be made, along with the reason(s) why no finding can be made. Because of their significance and distribution, Type I–III letters would be sent out under the signature of the NRC Chairman. While each of the three types of notification letters are important, the Type I letter is the most serious because it pertains to an immediate potential threat to public health and safety.

Table 4. Types of Notification Letters			
Type	**Notification**	**Signature**	**Distribution**
I	Indication that performance objective(s) are currently not being met	Chairman	DOE, covered State, and Congress
II	Indication that performance objective(s) will not be met in the future	Chairman	DOE, covered State, and Congress
III	Insufficient basis to determine that the performance objective(s) will be met	Chairman	DOE, covered State, and Congress
IV	Concerns with the performance demonstration	Staff	DOE and covered State
V	Resolution of concerns with the performance demonstration	Staff	DOE and covered State

Prior to sending out Type I–III letters, NRC will review its concerns in a letter (Type IV) to DOE and the State. This will give the State an opportunity to provide input and comments and DOE an opportunity to provide information that demonstrates its compliance with the performance objectives. Assuming that DOE provides information to support its performance demonstration, NRC will need to review this information and decide whether it is still a sufficient indication that the performance objectives are not or will not be met. If the staff determines that, based on the information provided by DOE, there is no longer a basis to conclude that DOE is noncompliant, NRC will send out a notification of resolution letter (Type V). Types IV and V letters will be made publicly available on NRC's website. These letters formally document issue resolution. If the staff determines that, based on the information DOE provides, there is still a basis for concluding that DOE is noncompliant, NRC will send out the notification of noncompliace letter (i.e., Type I–III).

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APPENDIX A

SUMMARY OF FACTORS IMPORTANT TO ASSESSING COMPLIANCE

Factor	Area	Description
1	Oxidation of Saltstone	The rate of waste oxidation is a key factor in the future performance of the saltstone disposal facility because the release of technetium is very dependent on the extent of oxidation of the saltstone wasteform. Realistic modeling of waste oxidation is needed to assure that the performance objectives of 10 CFR 61.41 will be met. Adequate model support is essential to providing the technical basis for the model results.
2	Hydraulic Isolation of Saltstone	To better understand the future performance of the disposal facility, it is important to understand the mechanisms of degradation of the wasteform to predict the rate of degradation as well as the expected physical properties of the degraded wasteform, such as hydraulic conductivity and diffusivity.
3	Model Support	Adequate model support is essential to assessing whether the saltstone disposal facility can meet 10 CFR 61.41. The model support for the following items is key to confirming the performance assessment results: (i) moisture flow through fractures in the concrete and saltstone located in the vadose zone, (ii) realistic modeling of waste oxidation and release of technetium, (iii) the extent and frequency of fractures in saltstone and vaults that will form over time, (iv) the plugging rate of the lower drainage layer of the engineered cap, and (v) the long-term performance of the engineering cap as an infiltration barrier.
4	Erosion Control Design	Implementation of an adequate erosion control design is important to ensuring that 10 CFR 61.42 can be met, because the erosion control barrier will help to maintain a thick layer of soil over the vaults, which reduces the potential for intrusion into the waste.

Factor	Area	Description
5	Infiltration Barrier Performance	The design and performance of the infiltration control system is important for ensuring that 10 CFR 61.41 can be met, because the release of contaminants from the saltstone to the groundwater is predicted to be sensitive to the amount of infiltration.
6	Feed Tank Sampling	Implementation of an adequate waste sampling plan is important to ensuring that 10 CFR 61.41 and 10 CFR 61.42 can be met, because it is necessary to confirm that the concentration of highly radioactive radionuclides (HRRs) in treated salt waste (or grout) is less than or equal to the concentration assumed in the waste determination.
7	Tank 48 Wasteform	The chemical composition of the salt waste in Tank 48 differs from the salt waste in other tanks because it contains a substantial amount of organic salts. To ensure that Tank 48 waste can be safely managed, tests are needed to measure the physical properties of the wasteform made from this waste to confirm that it will provide suitable performance.
8	Removal Efficiencies	The removal efficiencies of HRRs by each of the planned salt waste treatment processes are a key factor in determining the radiological inventory disposed of in saltstone, which, in turn, is an important factor in determining that 10 CFR 61.41 and 10 CFR 61.42 can be met.

APPENDIX B

SUMMARY OF MONITORING ACTIVITIES

Performance Objective	Chapter	Technical Review Activities	Onsite Observation Activities
61.41	3	• Review information on reported inventories and concentrations in the SDF. • Review groundwater monitoring data, updates to the monitoring plan, and quality assurance plans for sampling. • Factor 1 • Review information on vault design as it relates to oxidation. • Review information on gas phase transport of oxygen within the saltstone. • Review field and laboratory experiments, and any additional modeling of saltstone oxidation and Tc release. • Factor 2 • Review information to support the exclusion from consideration of specific saltstone degradation mechanisms. • Review information on curing technique and curing time for grout and concrete. • Review information on water condensation within the vaults. • Review information on the dissolution of salts and low solubility matrix phases within the grout.	• Observe any experiments performed to address issues related to Factor 3. • Observe waste sampling activities. • Observe vault construction and performance. • Review information on grout formulation and grout curing conditions. • Evaluate the adequacy of DOE's program for verifying the specifications of blast furnance slag. • Review DOE groundwater sampling process and installation of new wells.

Performance Objective	Chapter	Technical Review Activities	Onsite Observation Activities
61.41	3	• Factor 3 • Review any new moisture characteristic data for concrete and saltstone. • Review available information on the rate of equilibrium of water content within the saltstone. • Review any additional modeling analysis of moisture flow in the saltstone. • Review DOE conceptual model for oxidation and Tc release and any support for the model. • Review laboratory and field studies on concrete and saltstone cracking. • Review experiments and field studies that simulate processes related to plugging of the drainage layer through colloidal clay migration. • Review any experiments, analyses, or expert elicitation regarding the long-term performance of the infiltration barrier. • Factor 6 • Review DOE waste sampling plan and quality assurance procedures for sampling waste. • Review waste sampling data for the feed tank (Tank 50). • Factor 7 • Review DOE approach for treating waste in Tank 48. • Review characterization information for Tank 48. • Review information on the expected physical properties of the Tank 48 wasteform.	

Performance Objective	Chapter	Technical Review Activities	Onsite Observation Activities
61.41	3	• Factor 8 • Review information on radionuclide removal efficiencies by the various treatment processes. • Review estimates of the amount of sludge entrained in the salt waste during the DDA process. • Evaluate updates or revisions to DOE PA and special analysis.	
61.42	4	• Factor 4 • Evaluate technical details of the proposed closure cap. • Evaluate the design of erosion control features. • Evaluate updates or revisions to DOE intruder analysis.	
61.43	5	• Review reports related to worker and general public doses. • Review air effluent data from the salt waste processing facility. • Review information on DOE's quality assurance program for monitoring air emissions.	• Review DOE radiation protection program. • Observe DOE process for obtaining air effluent data.
61.44	6		• Observe the disposal facility for obvious signs of degeneration.

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