U.S. Nuclear Regulatory Commission Plan for Monitoring Disposal Actions Taken by the U.S. Department of Energy at the Savannah River Site Saltstone Disposal Facility in Accordance With the National Defense Authorization Act for Fiscal Year 2005, Revision 1

September 2013

U.S. Nuclear Regulatory Commission
Office of Federal and State Materials
and Environmental Management Programs
Washington, DC 20555-0001

(Intentionally Left Blank)

U.S. Nuclear Regulatory Commission Plan for Monitoring Disposal Actions Taken by the U.S. Department of Energy at the Savannah River Site Saltstone Disposal Facility in Accordance With the National Defense Authorization Act for Fiscal Year 2005, Revision 1

September 2013

U.S. Nuclear Regulatory Commission
Office of Federal and State Materials
and Environmental Management Programs
Washington, DC 20555-0001

(Intentionally Left Blank)

TABLE OF CONTENTS

FIGURES

TABLES

TABLE **PAGE**

ACRONYMS AND ABBREVIATIONS

Acronym or Abbreviation	Meaning
ALARA	As Low As (Is) Reasonably Achievable
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
DDA	Deliquification, Dissolution, and Adjustment
DOE	United States Department of Energy
EPA	United States Environmental Protection Agency
FDC	Future Disposal Cell
FEPs	Features, Events, and Processes
FFA	Federal Facility Agreement
FTF	F-Tank Farm or F-Area Tank Farm
FY	Fiscal Year
GCL	Geosynthetic Clay Liner
GCP	General Closure Plan
GSA	General Separations Area
HDPE	High Density Polyethylene
HTF	H-Tank Farm and H-Area Tank Farm
K_d	Distribution Coefficient
LLRW	Low-Level Radioactive Waste
MA	Monitoring Area
MCC	Moisture Characteristic Curve
MF	Monitoring Factor
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
NPO	Non-Compliant Performance Objective
NRC	United States Nuclear Regulatory Commission
PA	Performance Assessment
PMP	Probable Maximum Precipitation
PO	Performance Objective
RAI	[NRC] Request for Additional Information
SCDHEC	South Carolina Department of Health and Environmental Control
RG	[NRC] Regulatory Guide
SDF	Saltstone Disposal Facility
SDS	Saltstone Disposal Structure
SPF	Saltstone Production Facility
SRS	Savannah River Site
SWPF	Salt Waste Processing Facility
Tc	Technetium
TEDE	Total Effective Dose Equivalent
TER	Technical Evaluation Report
UTR	Upper Three Runs
WAC	Waste Acceptance Criteria
WD	Waste Determination
WIR	Waste Incidental to Reprocessing

DEFINITIONS

As Low As (Is) Reasonably Achievable (ALARA): From 10 CFR 20.1003 – Making every reasonable effort to maintain exposures to radiation as far below the dose limits as practical, consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health, and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy, and licensed materials in the public interest.

Closed: State of the land disposal facility when DOE does not dispose more radioactive waste in that facility.

Closed Monitoring Factor (MF): State of an MF after NRC staff determines and documents that the MF is no longer applicable or the technical issue or uncertainty has been resolved.

Controlled After Closure: State of the land disposal facility when DOE takes actions to restrict access to that facility after it has been closed.

Designed: State of the land disposal facility when DOE is calculating, planning, considering, operating, and closing the land disposal facility.

Disposal: The isolation of radioactive wastes from the biosphere.

Disposal Structure: In the context of the Saltstone Disposal Facility and, as discussed with DOE, NRC uses the term "disposal structure" to mean a self-enclosed entity used to contain saltstone and isolate it from the environment. A disposal structure could have more than one cell. The term "disposal structure" refers to the disposal entities that DOE refers to as "Vaults" and "Future Disposal Cells" (FDCs). In addition, the term "disposal structure" can be used to refer to any new potential disposal entity design that meets this definition. In the 2012 NRC Technical Evaluation Report, NRC used the term "disposal unit" to mean what is now meant by the term "disposal structure." DOE uses the designation Saltstone Disposal Unit (SDU) to denote something that contains one or more disposal structures. For example, the DOE designation of SDU 4, also referred to by DOE as Vault 4, is a disposal structure. For example, the DOE designation of SDU 2, also referred to by DOE as SDU 2A/2B, includes two disposal structures. The designation of a Saltstone Disposal Structure (SDS) will now be used to refer to a specific disposal structure (e.g., SDS 1, SDS 4, SDS 2A, SDS 2B, SDS 3A, SDS 3B).

DOE 2009 Performance Assessment (PA): This refers to the DOE 2009 PA and the two DOE responses to the NRC Requests for Additional Information (RAIs).

Follow-Up Action: Items identified during monitoring that require additional effort by DOE to resolve. Examples include DOE providing answers to questions generated during technical reviews or DOE providing the results of a particular experiment after it becomes available. A Follow-Up Action is less risk-significant than an Open Issue.

Highly Radioactive Radionuclide (also called Key Radionuclide): One of the radionuclides that contribute most significantly to risk to the public, workers, and the environment. In the context of the NDAA, NRC staff considers "Highly Radioactive Radionuclides," to be equivalent to "Key Radionuclides" used in the DOE Manual (DOE M 435.1-1) for DOE Order 435.1, the West Valley Policy Statement, and in some NRC reviews of DOE Waste Determinations (WDs). In the context of an NRC review of a DOE WD conducted under the NDAA, "Highly Radioactive Nuclides" are not (in general) limited to radionuclides with high specific activity.

Indeterminate: Insufficient information is currently available to assess compliance with the POs in 10 CFR Part 61, Subpart C. Additional information is forthcoming from DOE within a reasonable timeframe to allow NRC staff to assess compliance with the POs.

Land Disposal Facility: From 10 CFR 61.2 – The land, building, and structures, and equipment, which are intended to be used for the disposal of radioactive wastes.

Monitoring Area (MA): General feature or aspect of the disposal action identified by the NRC as being important to the DOE ability to meet the POs of 10 CFR Part 61, Subpart C. An MA is further divided into one or more specific Monitoring Factors (MFs).

Monitoring Factor (MF): A specific feature of the disposal action (e.g., conceptual model assumption, mathematical modeling assumption, parameter value) that DOE used in the demonstration that the NRC has determined to be important to demonstrating compliance with the POs of 10 CFR Part 61. NRC typically identifies an MF through the review of a DOE WD, PA, information that DOE generated during monitoring (e.g., technical report on laboratory or field experiment), or other information collected during monitoring (e.g., during NRC onsite observation visit). An MF is associated with an MA and tracked as either Open or Closed.

Non-Compliant Performance Objective (NPO): An NPO falls into one of the following categories:

- NPO-I: Evidence PO Not Met – NRC staff concludes that direct evidence (e.g., environmental sampling data) exists that indicates DOE disposal actions do not meet one or more POs in 10 CFR Part 61, Subpart C. Notification: NRC will issue a Type-I letter of non-compliance if DOE cannot demonstrate that executed disposal actions currently meet the requirements specified in the POs.

- NPO-II: Lack of Compliance Demonstration – NRC staff concludes that indirect evidence (e.g., experimental data on a key modeling assumption) exists that indicates DOE disposal actions do not meet one or more of the POs in 10 CFR Part 61, Subpart C. Notification: NRC will issue a Type-II letter of non-compliance if DOE cannot adequately address NRC technical concerns.

- NPO-III: Insufficient Information – NRC staff concludes that insufficient information is available to assess whether DOE disposal actions meet the POs in 10 CFR Part 61, Subpart C. It is not clear to NRC staff that either DOE has plans to or is able to provide the information in a reasonable timeframe to allow NRC staff to assess compliance. Notification: NRC will issue a Type-III letter of non-compliance if DOE cannot adequately address NRC technical concerns.

Onsite: Areas of the DOE site where monitoring activities will be carried out. This may include areas that have some relationship to, but are outside of, the physical boundaries of a particular WIR-related facility.

Onsite Observation Visit: A formal, pre-announced site visit to a DOE WIR-related facility by NRC staff for the purpose of observing DOE facilities, activities, processes, or experiments related to compliance with 10 CFR Part 61 POs.

Open Issue: A concern that NRC staff identifies during monitoring activities, which requires additional follow-up action or additional information from DOE to address regarding DOE disposal actions. Examples include an MF that DOE has not taken sufficient action to address or an instance where data collected by DOE are not consistent with assumptions (e.g., conceptual model assumptions, mathematical assumptions, parameter values) that DOE made in the PA. An Open Issue is more risk-significant than a Follow-Up Action. An Open Issue could lead to an NPO.

Operated: State of the land disposal facility during which time DOE carried out its waste disposal actions through the end of the institutional control period.

Operations: The actions taken by DOE while carrying out waste disposal actions through the end of the institutional control period, including PA development, waste removal, grouting, stabilization, observation, maintenance, and other similar activities.

Performance Assessment (PA): A type of systematic risk analysis that addresses the following four questions: (i) what can happen?, (ii) how likely is that to happen?, (iii) what are the resulting impacts of that happening?, and (iv) how do those impacts compare to specifically defined standards?

Performance Objective (PO): One of five 10 CFR Part 61, Subpart C, requirements for low-level waste disposal facilities, which are: (i) general requirement (§61.40), (ii) protection of the general population from releases of radioactivity (§61.41), (iii) protection of individuals from inadvertent intrusion (§61.42), (iv) protection of individuals during operations (§61.43), and (v) stability of the disposal site after closure (§61.44).

Reasonable: From the NRC Public Website Glossary – Rational, sensible, or resulting from sound judgment.

Sited: State of the land disposal facility when DOE performed actions to physically locate the facility.

Technical Review: NRC staff review of reports, studies, analyses, experiments, and other information prepared by DOE, SCDHEC, the public, stakeholders, or Native American Tribal Governments that may be used by NRC staff in determining the DOE ability to meet the 10 CFR Part 61 POs for NDAA waste disposal actions.

Waste Determination (or Non-High-Level Waste Determination): DOE documentation required by Section 3116 of the NDAA, which demonstrates that a specific waste stream is not high-level waste.

Worker: DOE or contractor staff who carry out operational activities at the land disposal facility.

EXECUTIVE SUMMARY

The Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA) authorizes the U.S. Department of Energy (DOE) in consultation with the U.S. Nuclear Regulatory Commission (NRC) to determine whether certain radioactive waste related to reprocessing of spent nuclear fuel is not high-level waste provided certain criteria are met. The NDAA applies specifically to DOE facilities in South Carolina and Idaho. The NDAA does not apply to similar DOE facilities located in other states. Section 3116(b) of the NDAA requires NRC to coordinate with the covered State (i.e., South Carolina or Idaho) to monitor DOE disposal actions to assess compliance with the five Performance Objectives (POs) for low-level waste in 10 CFR Part 61, Subpart C. The five POs are: (i) General Requirement (§61.40), (ii) protection of the general population from releases of radioactivity (§61.41), (iii) protection of individuals against inadvertent intrusion (§61.42), (iv) protection of individuals during operations (§61.43), and (v) stability of the disposal site after closure (§61.44). This monitoring plan details the NRC path forward in assessing the DOE compliance with the POs for the Saltstone Disposal Facility (SDF) at the Savannah River Site (SRS) near Aiken, South Carolina.

Based on the DOE 2005 Draft Waste Determination (WD), Performance Assessment (PA), and associated documents of the SDF, NRC staff concluded in the 2005 TER that the POs could be met if certain assumptions were verified via monitoring. Based on the NRC 2005 TER, NRC monitoring activities of the SDF began and currently follow the 2007 SDF Monitoring Plan, Rev. 0 *(NRC, 2007a).* After this document is issued, NRC will follow this 2013 SDF Monitoring Plan.

A PA is a type of systematic risk analysis that addresses the following four questions: (i) what can happen? (ii) how likely is that to happen?, (iii) what are the resulting impacts of that happening?, and (iv) how do those impacts compare to specifically defined standards? Considering the long time period that long-lived radionuclides could present a risk to human health, a robust PA is needed to establish that the POs will be met for releases from the SDF that may occur many thousands of years in the future. NRC considers sufficient DOE PA model support with NRC observation of disposal actions necessary for NRC to have reasonable assurance that the POs can be met.

DOE revised their 2005 PA in 2009. NRC used the DOE 2009 PA along with the DOE responses to the two NRC Requests for Additional Information (RAIs) to assess compliance with three of the five POs (i.e., §61.41, §61.42, §61.44). NRC used other monitoring activities to assess compliance with §61.43. On April 30, 2012, NRC issued the Technical Evaluation Report (TER) *(NRC, 2012)*. In the 2012 TER, the NRC concluded that it did not have reasonable assurance that DOE salt waste disposal at the SDF would meet the POs, specifically §61.41. The NRC also concluded that it did have reasonable assurance that DOE will meet §61.42, §61.43, and §61.44. The 2012 TER did not explicitly indicate whether or not the NRC concluded with reasonable assurance that DOE will meet §61.40, but because the NRC concluded it did not have reasonable assurance that DOE disposal actions would meet §61.41, then the NRC could not have reasonable assurance that DOE disposal actions would meet §61.40.

To prepare this monitoring plan, the NRC staff considered the “Factors” from Appendix A of the NRC 2012 TER and the three current “Open Issues” under the NRC 2007 Monitoring Plan. NRC linked those concerns (i.e., Factors, Open Issues) to one of the MAs with associated MFs that the NRC staff determined was important to the DOE compliance demonstration.

NRC monitoring activities to assess DOE compliance with 10 CFR 61.40, “General Requirements,” will not be evaluated separately from the other POs because it is the general requirement that exposures to humans are within the limits established in the performance objectives in §61.41, §61.42, §61.43, and §61.44. If the NRC concludes with reasonable assurance that DOE complies with §61.41, §61.42, §61.43, and §61.44, then NRC will also conclude with reasonable assurance that DOE complies with §61.40. Otherwise, NRC cannot conclude with reasonable assurance that DOE complies with §61.40.

NRC monitoring activities to assess DOE compliance with §61.41, “Protection of the General Population from Releases of Radioactivity,” will be evaluated through a risk-informed review, including onsite observation visits, technical reviews, and data reviews, of MA 1 through MA 10. The listing of those 10 MAs is below:

- MA 1 Inventory
- MA 2 Infiltration and Erosion Control
- MA 3 Waste Form Hydraulic Performance
- MA 4 Waste Form Physical Degradation
- MA 5 Waste Form Chemical Degradation
- MA 6 Disposal Structure Performance
- MA 7 Subsurface Transport
- MA 8 Environmental Monitoring
- MA 9 Site Stability
- MA 10 Performance Assessment Model Revisions

The associated MFs under MA 1 through MA 9 are listed below:

- MF 1.01 Inventory in Disposal Structures
- MF 1.02 Methods Used to Assess Inventory
- MF 2.01 Hydraulic Performance of Closure Cap
- MF 2.02 Erosion Protection
- MF 3.01 Hydraulic Conductivity of Field-Emplaced Saltstone
- MF 3.02 Variability of Field-Emplaced Saltstone
- MF 3.03 Applicability of Laboratory Data to Field-Emplaced Saltstone
- MF 3.04 Effect of Curing Temperature on Saltstone Hydraulic Properties
- MF 4.01 Waste Form Matrix Degradation
- MF 4.02 Waste Form Macroscopic Fracturing
- MF 5.01 Radionuclide Release from Field-Emplaced Saltstone
- MF 5.02 Chemical Reduction of Tc by Saltstone
- MF 5.03 Reducing Capacity of Saltstone
- MF 5.04 Certain Risk-Significant K_d Values for Saltstone
- MF 5.05 Potential for Short-Term Rinse-Release from Saltstone
- MF 6.01 Certain Risk-Significant K_d Values in Disposal Structure Concrete
- MF 6.02 Tc Sorption in Disposal Structure Concrete
- MF 6.03 Performance of Disposal Structure Roofs and HDPE/GCL Layers
- MF 6.04 Disposal Structure Concrete Fracturing
- MF 6.05 Integrity of Non-cementitious Materials
- MF 7.01 Certain Risk-Significant K_d Values in Site Sand and Clay
- MF 8.01 Leak Detection
- MF 8.02 Groundwater Monitoring

- MF 9.01 Settlement Due to Increased Overburden
- MF 9.02 Settlement Due to Dissolution of Calcareous Sediment

When reviewing the DOE 2009 PA and associated RAI responses, the NRC staff identified concerns that it expects will not be resolved until DOE revises the PA model under the DOE PA maintenance program. Those items are identified as MFs 10.01 through 10.13, as listed below:

- MF 10.01 Implementation of Conceptual Models
- MF 10.02 Defensibility of Conceptual Models
- MF 10.03 Diffusivity in Degraded Saltstone
- MF 10.04 K_d Values for Saltstone
- MF 10.05 Moisture Characteristic Curves
- MF 10.06 K_d Values for Disposal Structure Concrete
- MF 10.07 Calculation of Build-Up in Biosphere Soil
- MF 10.08 Consumption Factors and Uncertainty Distributions for Transfer Factors
- MF 10.09 K_d Values for SRS Soil
- MF 10.10 Far-Field Model Calibration
- MF 10.11 Far-Field Model Source Loading Approach
- MF 10.12 Far-Field Model Dispersion
- MF 10.13 Impact of Calcareous Zones on Contaminant Flow and Transport

NRC monitoring activities to assess DOE compliance with §61.42, "Protection of Individuals from Inadvertent Intrusion," will be evaluated through a risk-informed review, including onsite observation visits, technical reviews, and data reviews, of MA 1 through MA 10, which are the same as those MAs listed above under §61.41. The associated MFs applicable to §61.41 are also applicable to §61.42. However, MF 10.10 – MF 10.13 do not apply to §61.42.

NRC monitoring activities to assess DOE compliance with §61.43, "Protection of Individuals During Operations," will be evaluated through a risk-informed review, including onsite observation visits, technical reviews, and data reviews, of the DOE radiological protection program. NRC staff will review radiation records, review environmental data, review environmental reports, and conduct interviews to assess compliance with §61.43 under MA 11 (Radiation Protection Program). Specifically, those activities are under MF 11.01 (Dose to Individuals During Operations) and MF 11.02 (Air Monitoring).

NRC monitoring activities to assess DOE compliance with §61.44, "Stability of the Disposal Site After Closure," will be evaluated through a risk-informed review, including onsite observation visits, technical reviews, and data reviews, and partially overlap the NRC monitoring activities to assess DOE compliance with §61.41 and §61.42. The unique aspects affecting stability of the disposal site after closure that are not under §61.41 and §61.42 will be reviewed under MA 9 (Site Stability); specifically, MF 9.01 (Settlement Due to Increased Overburden) and MF 9.02 (Settlement Due to Dissolution of Calcareous Sediment).

This monitoring plan will be used by NRC staff to continue to assess DOE compliance with the POs in 10 CFR Part 61, Subpart C, in fulfillment of the NRC monitoring responsibilities under the NDAA. Each MA is supported by one or more MF. Each MF is a smaller, more specific item that NRC staff will monitor in more detail. Each MF will be tracked as Open or Closed. If NRC staff concerns arise related to an MF, then NRC staff may develop an "Open Issue" to document concerns related to that MF. Thus, NRC will have a mechanism to communicate to DOE early of the need for corrective action, prior to issuance of a Notification Letter.

This monitoring plan is organized by PO with a chapter devoted to each of the five POs. Each chapter has the associated MA and its associated MF. Each MA supports one or more POs. If the MA supports multiple POs, then this monitoring plan indicates whether or not the MA with associated MFs are an exact duplicate of a previously listed MA (i.e., whereby the details of the MA and MFs will not be repeated) or if there are unique aspects of the MA or associated MFs that pertain to that PO. At the end of each MF, there is a specific paragraph that provides the current NRC expectations of how and/or when NRC expects that the MF will be closed. **When this monitoring plan mentions the DOE 2009 PA, this refers to the DOE 2009 PA and the two DOE responses to the NRC RAIs.**

(Intentionally Left Blank)

1. INTRODUCTION

1.1. Background

SRS is a 780 square kilometer (310 square mile) facility located in south-central South Carolina, which began operation in 1951 producing nuclear materials for national defense, research, medical, and space programs. Waste produced at the site from spent nuclear fuel reprocessing for defense purposes has been commingled with non-reprocessing waste resulting from the production of targets for nuclear weapons and production of material for space missions. Significant quantities of radioactive waste are currently stored onsite in large underground waste storage tanks, which were placed in operation between 1954 and 1986. The waste stored in the tanks at SRS is a mixture of insoluble metal hyrdroxide solids, referred to as sludge, and soluble salt supernate. The supernate volume has been reduced by evaporation, which also concentrates the soluble salts to their solubility limits. The resultant solution crystallizes as salts and the resulting solid is referred to as saltcake. The saltcake and supernate combined are referred to as salt waste. DOE removes the salt waste, treats it to remove highly radioactive radionuclides to the maximum extent practical, and disposes of the low activity fraction onsite in the SDF. The SDF is located in the Z-area of the SRS, which is approximately 10 kilometers (6.2 miles) from the nearest SRS site boundary on a well-drained local topographic high.

NRC documented the review of the DOE 2005 WD, PA, and associated documents in the NRC 2005 TER *(NRC, 2005)*. The DOE SDF Final Waste Determination *(DOE, 2006)*, included the following: "Section 3116 of the NDAA authorizes the Secretary of Energy, in consultation with the NRC, to determine that certain waste from reprocessing is not high-level waste if it meets the criteria set forth in that Section: (1) that it does not require disposal in a deep geologic repository, (2) that it has had highly radioactive radionuclides removed to the maximum extent practical, (3) that it meets concentration limits and/or dose-based performance objectives for near-surface disposal of radioactive waste, and (4) that it will be disposed of pursuant to a State-issued permit or State-approved closure plan. In this document, the Secretary is determining that the treated, solidified low-activity salt waste from the tanks meets all of those criteria. Accordingly, the Secretary is determining that this material is not high-level waste and will be disposed in SDF."

NRC monitoring activities of the SDF began in 2006 and currently follow the 2007 SDF Monitoring Plan, Rev. 0 *(NRC, 2007a).* Those activities included onsite observation visits, technical reviews, data reviews, and meetings with DOE. On April 30, 2012, the NRC issued the Technical Evaluation Report (TER) *(NRC, 2012)*. In the 2012 TER, the NRC concluded that it did not have reasonable assurance that salt waste disposal at the SDF will meet the POs, specifically, §61.41. The 2012 TER did not explicitly indicate whether or not the NRC concluded with reasonable assurance that DOE will meet §61.40, but because the NRC concluded it did not have reasonable assurance DOE disposal actions would meet §61.41, the NRC could not have reasonable assurance DOE disposal actions would meet §61.40.

1.2. Objective

In accordance with Section 3116 of the NDAA, after the Secretary of Energy has made a determination that the residual waste is not high-level waste, then NRC is required, in coordination with the SCDHEC, to monitor subsequent DOE disposal actions to assess compliance with the POs. This plan describes those activities that NRC and South Carolina believe to be important to assess compliance with the POs. The DOE actions associated with the SDF will take decades to complete and NRC implementation of this monitoring plan will take place concurrently with those DOE activities. NRC monitoring activities include the following:

- Technical reviews of DOE work products, experiments, and analyses tied to one or more MAs, including collection of environmental data.
- Data reviews of information that DOE has collected (e.g., radiation records).
- Periodic (i.e., quarterly or less frequently) onsite observation visits of aspects of DOE disposal activities and, as appropriate, related experiments.

NRC monitoring activities will be accomplished by NRC headquarters and regional staff. NRC staff will coordinate with SCDHEC staff regarding accomplishment of monitoring tasks supportive of both programs.

1.3. Roles and Responsibilities

1.3.1. Federal Facility Agreement Roles and Responsibilities for SRS

The SRS Federal Facility Agreement (FFA is a formal agreement between the DOE, the U.S. Environmental Protection Agency (EPA), and the State of South Carolina with only those three entities having roles and responsibilities under the FFA. EPA is a party to the FFA pursuant to its authority in accordance with the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), also known as Superfund, under which EPA has been tasked to protect citizens from the dangers posed by abandoned or uncontrolled hazardous wastes. While Congress writes environmental laws, EPA writes regulations to implement those laws, enforces the regulations, and sets national standards. The EPA role in the FFA is to provide oversight of any actions taken at SRS to ensure adherence with CERCLA, the FFA, and other guidance. EPA is also responsible for providing technical and procedural assistance, information, and training as needed. More specifically, the EPA involvement with the State is focused on ensuring that proper disposal actions are taken, assisting the State with the design and installation of those actions, and monitoring and evaluating the effectiveness of those actions. Executive Order 12580 delegates the responsibility to implement the provisions in CERCLA to DOE and the U.S. Department of Defense. CERCLA also names DOE and the U.S. Department of Defense as the lead agencies for their respective areas. DOE has several facilities in EPA Region 4. SRS was added to the Superfund National Priorities List in December of 1989, which was also the year that SRS was required to have an FFA with the State of South Carolina and EPA. SCDHEC is the primary regulator of DOE closure activities at SRS. The State of South Carolina has authority for approval of wastewater treatment facility operational closure under Chapter 61, Articles 67 and 82 of the SCDHEC Regulations (SCDHEC R.61-67, SCDHEC R.61-82).

1.3.2. DOE and NRC NDAA Monitoring Roles and Responsibilities

Under the NDAA, only the DOE, NRC, and covered States, including South Carolina for the SRS, have roles and responsibilities.

DOE will, pursuant to its authority, perform saltstone disposal actions and monitor its activities to ensure compliance with all requirements. DOE's relevant authority stems from the Atomic Energy Act of 1954, as amended, and applicable DOE orders, manuals, and policies. Furthermore, DOE uses a documented process to review and resolve any disposal questions and develop any mitigation measures, as appropriate. The DOE roles and responsibilities under NDAA with respect to monitoring are to respond to NRC and SCDHEC monitoring activities. Examples include the following: (i) provide fact check input on NRC monitoring plans, NRC observation guidance memoranda, NRC onsite observation visit reports, (ii) participate in NRC onsite observation visits, teleconference calls, and meetings, and (iii) in a timely manner, communicate to NRC any concerns about DOE monitoring actions.

Section 3116 of the NDAA authorized the Secretary of Energy to manage and dispose of certain waste associated with facility clean-up in Idaho and South Carolina, as low-level waste, in accordance with the POs in NRC regulations. Prior to such determination, DOE is required to consult with NRC regarding its waste determination. Following the Secretary of Energy's approval of the final waste determination, the NRC is required to monitor DOE disposal actions in coordination with the covered State.

The NRC's role in monitoring DOE's disposal actions derives from Section 3116 of the NDAA. While the NRC is not given a formal regulatory role, the NDAA requires that NRC monitor, in coordination with SCDHEC, DOE disposal actions to assess compliance with the POs in 10 CFR Part 61. Thus, NRC assesses DOE's compliance with a subset of NRC regulations in 10 CFR Part 61, "Licensing Requirements Land Disposal of Radioactive Waste" in carrying out such disposal procedures, criteria, and terms and conditions upon which NRC issues licenses for the disposal of radioactive wastes containing byproduct, source, and special nuclear material received from other persons.

This monitoring plan articulates NRC's role to assess, in coordination with SCDHEC, DOE's disposal actions associated with residual waste covered by the DOE Secretary's approval of the final waste determination regarding compliance with the POs in 10 CFR Part 61. Examples include the following: (i) draft and finalize NRC monitoring plans, incorporating SCDHEC input and DOE fact check input; (ii) draft and finalize NRC observation guidance memoranda for NRC onsite observation visits, including incorporating SCDHEC input and DOE fact check input; (iii) perform NRC onsite observation visits; (iv) draft and finalize NRC onsite observation visit reports, incorporating SCDHEC input and DOE fact check input; (v) perform NRC technical reviews and NRC data reviews; (vi) draft and finalize NRC reports for NRC technical reviews and NRC data reviews; (vii) lead teleconference calls and meetings; (viii) in a timely manner, communicate to DOE and SCDHEC any NRC concerns about DOE not meeting the POs; and (ix) notify Congress if NRC determines that DOE is not in compliance with the POs.

1.4. Coordination with the State of South Carolina

SCDHEC is the primary regulator of site disposal activities and a key part of the NRC monitoring responsibility under the NDAA, which requires NRC to coordinate all monitoring activities with the covered State. During the 2005 review, NRC staff began coordinating with the State of South Carolina by conducting discussions to determine the types of activities that the State will undertake to monitor compliance with its regulatory requirements. Those discussions enabled the State to gain a better understanding of NRC activities. NRC continues to coordinate with the SCDHEC throughout the monitoring process by consulting with SCDHEC in developing future revisions to this monitoring plan and reviewing the State's Environmental Surveillance and Oversight Program as a source of information to supplement the NRC monitoring plan. SCDHEC uses a holistic monitoring approach with regard to the overall performance and safety of SRS. The NRC objective with this NDAA monitoring program is to assess the compliance of a very small portion of the site with the five POs. Ultimately, NRC and SCDHEC are concerned with the potential for environmental contamination in groundwater, surface water, air, crops, milk, and meat. While it is unlikely that any contribution to such contamination from the SDF could manifest itself offsite in the foreseeable future, it is important to consider and evaluate, to the extent practicable, the utility of environmental monitoring in assessing compliance with the

POs. To the extent that SCDHEC resources allow, NRC will request assistance in following up on certain monitoring activities that require a local or on-site presence.

The SCDHEC roles and responsibilities under NDAA with respect to monitoring are to perform activities under the South Carolina Environmental Surveillance and Oversight Program and to coordinate with NRC. Examples of those include the following: (i) provide input to NRC monitoring plans, NRC observation guidance memoranda; NRC onsite observation visit reports; (ii) participate in NRC onsite observation visits; (iii) as requested by NRC, perform technical reviews and data reviews and provide input to NRC reports for those reviews; (iv) in a timely manner, communicate to DOE and NRC any concerns about DOE not meeting the POs.

NRC will keep the State informed of the status of monitoring activities at the site, including any potential findings of DOE's non-compliance that require a notification letter to Congress. At least two business days prior to the release and dissemination of a notification letter, SCDHEC will be briefed on the reason(s) for the notification.

1.5. The Monitoring Process

Monitoring is an on-going process consisting of technical reviews, data reviews, and periodic (i.e., quarterly or less frequently) onsite observation visits of DOE disposal activities related to NRC assessing DOE's compliance with the 10 CFR Part 61 POs. NRC uses the information gathered from DOE and other publicly available sources to continuously assess DOE's compliance. Many of the DOE documents, including the PA and environmental monitoring reports, are written to satisfy DOE internal requirements or other regulatory requirements for DOE's regulators (i.e., DOE, EPA, SCDHEC). NRC will continue to leverage those already created documents and analyses to assist NRC in assessing DOE's compliance with the POs.

1.5.1. Technical Reviews

Technical reviews by NRC (or SCDHEC) include the review and evaluation of analyses conducted by DOE or others that address one or more aspects of site performance. Also, the reviews are used to assess model support for assumptions made by DOE in the PA that are considered important to the DOE compliance demonstration. NRC will document each review, which will be publicly available (e.g., Note to File, onsite observation visit report).

1.5.2. Data Reviews

Data reviews focus on real-time monitoring data that may indicate future system performance or a review of records or reports that can be used to directly assess compliance with POs (e.g., review of radiation records). NRC will document each review, which will be publicly available (e.g., Note to File, onsite observation visit report).

1.5.3. Onsite Observation Visits

Onsite observation visits are first-hand observations by NRC staff or a representative of a specific activity that could be used to assess an aspect of current or future site performance. A visit is generally performed to either: (1) ensure that data collected for a technical review are of sufficient quality; or (2) observe key disposal actions that are important to DOE's compliance demonstration (e.g., SDF operations, waste characterization).

Prior to each onsite observation visit, NRC will prepare an observation guidance memorandum that discusses the scope and specific activities that will be monitored during the visit in more detail than is described in this monitoring plan. The activities will be based on many aspects, such as completion of DOE technical reports, emergent issues, timely DOE actions related to an MF, availability of staff (i.e., NRC, SCDHEC, DOE), availability of locations at the site, length of time since reviewing an item in an MF, scheduled follow-up actions to previous visits, and available NRC resources. NRC will coordinate with SCDHEC in development of the memorandum to take into account areas that SCDHEC is interested in and availability of SCDHEC experts in those areas of interest. NRC plans to provide the final memorandum to DOE by 30 calendar days prior to the visit. The final memorandum will be publicly available. During a visit, the agenda may change based on what happens during the visit (e.g., new areas of interest are identified) or unforeseen circumstances (e.g., weather).

Each visit will be documented in an onsite observation visit report. The report will include, for the actual areas covered during the visit (i.e., may not be the same as the areas of interest identified in the observation guidance memorandum), specific activities, results of discussions, status of any Open Issues/Follow-Up Actions, and any NRC conclusions. NRC plans to finalize each report by 60 calendar days after the visit. The final report will be publicly available.

1.5.4. Periodic DOE Documents used in NRC Monitoring and Examples of Upcoming DOE Activities/Events

The DOE documents, including the PA and environmental monitoring, are written for DOE or other regulatory requirements for DOE's regulators (i.e., DOE, EPA, SCDHEC) and NRC is leveraging those already created documents and analyses to assist NRC in assessing DOE's compliance with the POs. At SDF, NRC has been performing monitoring activities since 2006. Based on that experience, both NRC and DOE agreed that in the future it would be beneficial to both federal agencies if DOE were to provide to NRC on a timely basis the following two sets of information:

- Certain original and updated DOE documents that are relevant to the NRC in performing its monitoring responsibilities (e.g., documents may be reviewed by NRC at NRC or during an onsite observation visit) that NRC would periodically review.
- Timely notification that certain upcoming DOE activities will be occurring in the near- or long-term future (i.e., no surprises to NRC, allowing enough time for NRC to understand the DOE activities and let DOE know if NRC has any concerns about those activities). In addition, timely notification when certain events happen.

Table 1-1 lists the periodic topical area, specific technical area, and DOE expected availability of periodic documents used in NRC monitoring activities.

Table 1-1: List of Periodic DOE Documents used in NRC Monitoring

Topical Area	Document	Approximate Availability/Frequency
Groundwater	* SDF Annual Groundwater Report * SDF Midyear Groundwater Report * SDF Performance Assessment Annual Review * SRS Annual Environmental Report	January July March September
Air Monitoring	* SRS Annual Environmental Report	September

Topical Area	Document	Approximate Availability/Frequency
Inventory	* SDF Performance Assessment Annual Review and Key Supporting Inventory References * Saltstone Permit Website Reporting Data (http://sro.srs.gov.saltstone.htm) * Salt Batch Qualification Reports# * Tank 50 WAC Sample Analysis	March Quarterly As issued Quarterly
Performance Assessment Maintenance	* SRS Liquid Waste Facilities Performance Assessment Maintenance Annual Implementation Plan * SDF Performance Assessment Annual Review (includes the following): - Inventory - Unreviewed Waste Management Question Evaluations Performed - Research and Development Performed - Research and Development Planned	March March
Research and Development Testing/Studies	Various reports	As issued (typically, September through December)
# *Note: DOE anticipates providing NRC with Salt Batch Qualification Reports throughout the timeperiod during which DOE has imposed lower Tc-99 limits on the salt solution that can be processed at SPF*		

Table 1-2 provides examples of categories with specific activities or events that NRC would like DOE to keep NRC informed about.

Table 1-2: Examples of Upcoming DOE Activities/Events

Category	Activity/Event
General	* DOE determines that a special analysis will be performed * DOE starts saltstone disposal into a new disposal structure * Major work stoppages or unusual events occur (e.g., spills, disposal structure wall seeps) * Significant changes are made to the salt waste treatment processes that could affect treated salt waste composition (e.g., radionuclide inventories, aluminate concentration) * SWPF operations begin
Inventory	* Changes are made to the methods used to assess the inventory of radionuclides disposed of at the SDF * Significant changes are made to the estimated inventory of Tc-99, Ra-226, Th-230, I-129, or Se-79 in disposal structures
Waste Form	Changes are made to the saltstone formulation, including changes to admixtures
Disposal Structure Performance	* DOE decides to use a new disposal structure design or significant changes are made to a disposal structure design * Major delays or unexpected events (e.g., unexpected hydrotest results) occur during disposal structure construction

1.6. Periodic Compliance Monitoring Report

NRC will publish a Periodic Compliance Monitoring Report (i.e., currently, annually as NUREG-1911) to document the major findings associated with the monitoring activities during each calendar year. The report will be for the entire NRC NDAA program for that calendar year and will be publicly available.

1.7. Notification Letters

There are five types of notification letters. Three of the letters are non-compliance letters (i.e., Type-I, Type-II, Type-III) that NRC developed to implement the authority it has inferred from the statutory language in Section 3116 of the NDAA and two of the letters (i.e., Type-IV, Type-V) are other letters that NRC may issue as an interim step, based on the NRC guidance in NUREG -1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations" *(NRC, 2007b)*.

The NRC expects to issue a Letter of Concern (i.e., Type-IV) to allow DOE sufficient time to respond to NRC concerns prior to issuance of one of the three non-compliant notification letters listed above (i.e., Type-I, Type-II, Type-III). However, that may not be possible or appropriate in all situations. For example, if a worker were to be overexposed in an accident (i.e., received greater than 5 rem exposure) and therefore, the NRC was going to issue a Type-I Letter of Non-Compliance, then the NRC may decide to send that notification letter to Congress, DOE, and the covered State rather than first sending a Type-IV Letter of Concern to DOE and the covered State. The NRC would utilize other means of notification (e.g., telephone conference calls or meetings) with both DOE and the covered State before sending the Letter of Non-Compliance.

Table 1-3 describes each type of notification letter, including the NRC conclusion for issuing the letter, the NRC basis for issuing the letter, who at the NRC signs the letter, and who receives the letter.

Table 1-3: Types of Notification Letters

Type	Notification	Signature	Distribution
Non-Compliant PO Notification Letters			
I	<u>Evidence PO Is Not Met:</u> NRC staff concludes that direct evidence (e.g., environmental sampling data) exists that indicates DOE disposal actions do not meet one or more POs in 10 CFR Part 61, Subpart C. Notification: NRC will issue a Type-I letter of non-compliance if DOE cannot demonstrate that executed disposal actions currently meet the requirements specified in the POs.	NRC Chairman	DOE, covered State, and Congress
II	<u>Lack of Compliance Demonstration:</u> NRC staff concludes that indirect evidence (e.g., data regarding key modeling assumptions) exists that indicates DOE disposal actions do not meet one or more of the POs in 10 CFR Part 61, Subpart C. Notification: NRC will issue a Type-II letter of non-compliance if DOE cannot adequately address NRC technical concerns.	NRC Chairman	DOE, covered State, and Congress

III	Insufficient Information: NRC staff concludes that insufficient information is available to assess whether DOE disposal actions meet the POs in 10 CFR Part 61, Subpart C. It is not clear to NRC staff that either DOE: (i) has plans to; or (ii) is able to provide the information in a reasonable timeframe to allow NRC staff to assess compliance. Notification: NRC will issue a Type -III letter of non-compliance if DOE cannot adequately address NRC technical concerns.	NRC Chairman	DOE, covered State, and Congress
"Other" Notification Letters			
IV	Concern: NRC staff concludes that there are concerns with DOE's demonstration of meeting the POs in 10 CFR Part 61, Subpart C. Notification: NRC will issue a Type-IV letter of concern if DOE cannot adequately address the NRC concerns.	NRC staff or NRC management	DOE and covered State
V	Resolution: NRC staff concludes that DOE has provided sufficient information to resolve the concerns in the Type-IV letter of concern regarding DOE's demonstration of meeting the POs in 10 CFR Part 61, Subpart C. Notification: NRC will issue a Type-V letter of resolution if DOE adequately addresses the NRC concerns in a Type IV letter of concern.	NRC staff or NRC management	DOE and covered State
Note: NRC expects to issue a Letter of Concern (i.e., Type-IV) to allow DOE sufficient time to respond to NRC staff concerns prior to issuance of one of the three notification letters of non-compliance (i.e., Type-I, Type II, or Type III) listed above.			

Figure 1-1 shows the types of non-compliance with the POs in 10 CFR Pat 61, Subpart C.

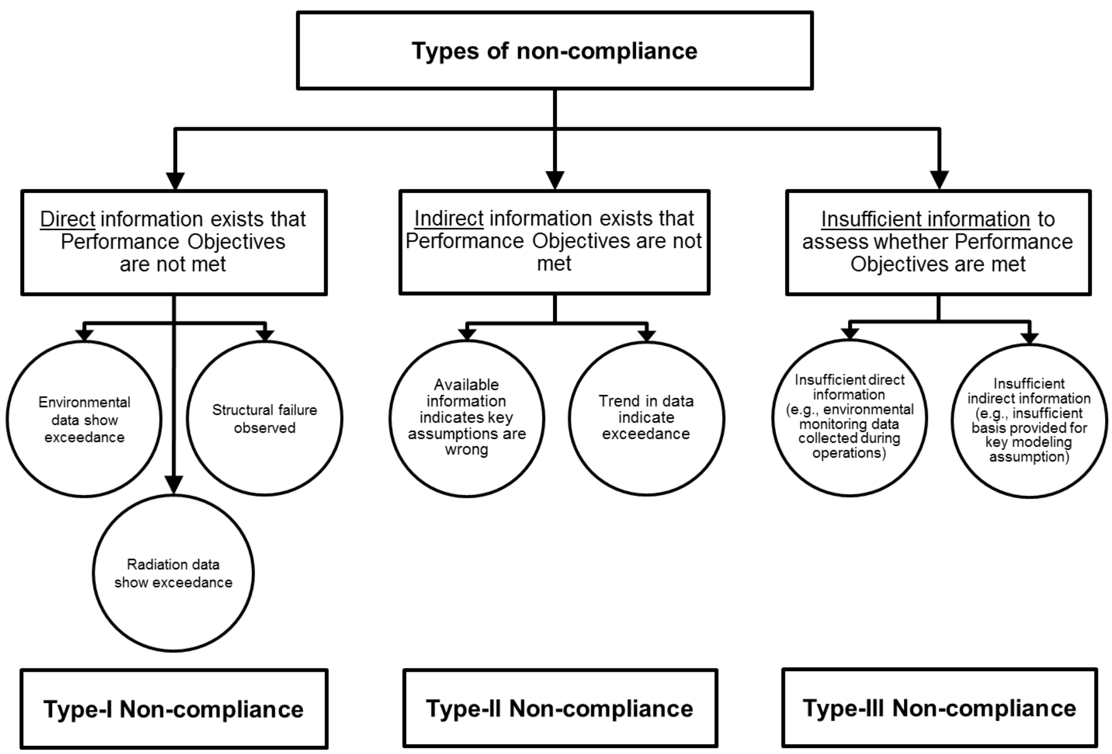

Figure 1-1: Types of Non-Compliance with the POs in 10 CFR Part 61, Subpart C

1.8. The Monitoring Plan

The NRC will be performing monitoring activities under this monitoring plan. The organization of this monitoring plan is by each of the five POs; within each PO is each appropriate MA; and within each MA is each appropriate MF. New MAs are not expected in the future, but they may be identified and added to a future revision of this monitoring plan.

The MFs were created from the concerns identified in the NRC 2012 TER *(NRC, 2012)*, which included the previous Open Issues under the 2007 Monitoring Plan. Each of the previous Open Issues is now considered to be administratively closed, but NRC still has each of those concerns described in the previous Open Issues, so those concerns will now be addressed under the MFs in this monitoring plan (see Section 1.8.1 – Link Between Previous Open Issues and Monitoring Factors).

The identification, description, and status (i.e., Open or Closed) of each of the MFs will evolve as monitoring activities continue in the future. New MFs are expected to be added to the monitoring plan in the future as more information is known about the future DOE disposal actions and experiments. After each onsite observation visit, NRC will issue a report that will document the status of each open item (i.e., MF, Open Issue, Follow-Up Action) and indicate which MFs were observed during that onsite observation visit. NRC expects to issue revisions to the monitoring plan in the future to address such things as, the results of an updated DOE PA or a new NRC TER, in a risk-informed, performance-based manner.

Table 1-4 provides the link between each PO and which of the MAs support that PO.

Table 1-4: Link between Monitoring Activities and Performance Objectives

MA#	Description of Monitoring Area	Performance Objective			
		§61.41	§61.42	§61.43	§61.44
1	Inventory	X	X		
2	Infiltration and Erosion Control	X	X		
3	Waste Form Hydraulic Performance	X	X		
4	Waste Form Physical Degradation	X	X		
5	Waste Form Chemical Degradation	X	X		
6	Disposal Structure Performance	X	X		
7	Subsurface Transport	X	X		
8	Environmental Monitoring	X	X		
9	Site Stability	X	X		X
10	Performance Assessment Model Revisions	X	X		
11	Radiation Protection Program			X	

Table 1-5 provides the link between the Sections in the NRC 2012 TER and which MFs came from those Sections. The information in Table 1-5 is primarily from the NRC 2012 TER Table A-1, “Key Monitoring Factors,” but with slight changes because the MFs in this monitoring plan are slightly different than the “Factors” in the NRC 2012 TER Table.

Table 1-5: Link Between 2012 TER Sections and Monitoring Factors

2012 TER Section	MF #	Description of Monitoring Factor
2.2	1.01	Inventory in Disposal Structures
2.2	1.02	Methods Used to Assess Inventory
2.4, 2.5, and 5	2.01	Hydraulic Performance of Closure Cap

2012 TER Section	MF #	Description of Monitoring Factor
2.4	2.01	Erosion Protection
2.6	3.01	Hydraulic Conductivity of Field-Emplaced Saltstone
2.6	3.02	Variability of Field-Emplaced Saltstone
2.6	3.03	Applicability of Laboratory Data to Field-Emplaced Saltstone
2.6	3.04	Effect of Curing Temperature on Saltstone Hydraulic Properties
2.6	4.01	Waste Form Matrix Degradation
2.6	4.02	Waste Form Macroscopic Fracturing
2.6	5.01	Radionuclide Release from Field-Emplaced Saltstone
2.6	5.02	Chemical Reduction of Tc by Saltstone
2.6 and 2.7	5.03	Reducing Capacity of Saltstone
2.6 and 2.7	5.04	Certain Risk-Significant K_d Values for Saltstone
2.7	5.05	Potential for Short-Term Rinse-Release from Saltstone
2.5 and 2.7	6.01	Certain Risk-Significant K_d Values in Disposal Structure Concrete
2.7 and 2.13	6.02	Tc Sorption in Disposal Structure Concrete
2.5 and 2.7	6.03	Performance of Disposal Structure Roofs and HDPE/GCL Layers
2.5	6.04	Disposal Structure Concrete Fracturing
2.5	6.05	Integrity of Non-cementitious Materials
2.7	7.01	Certain Risk-Significant K_d Values in Site Sand and Clay
* Not Applicable	8.01	Leak Detection
* Not Applicable	8.02	Groundwater Monitoring
2.4, 2.5, 2.6, and 5	9.01	Settlement Due to Increased Overburden
2.4, 2.5, 2.6, and 5	9.02	Settlement Due to Dissolution of Calcareous Sediment
2.11, 2.13, and 3	10.01	Implementation of Conceptual Models
2.13 and 3	10.02	Defensibility of Conceptual Models
2.6	10.03	Diffusivity in Degraded Saltstone
2.7	10.04	K_d Values for Saltstone
2.7	10.05	Moisture Characteristic Curves
2.7 and 2.13	10.06	K_d Values for Disposal Structure Concrete
2.10	10.07	Calculation of Build-Up in Biosphere Soil
2.10	10.08	Consumption Factors and Uncertainty Distributions for Transfer Factors
2.7	10.09	K_d Values for SRS Soil
2.8	10.10	Far-Field Model Calibration
2.8	10.11	Far-Field Model Source Loading Approach
2.8	10.12	Far-Field Model Dispersion
2.8	10.13	Impact of Calcareous Zones on Contaminant Flow and Transport
4	11.01	Dose to Individuals During Operations
4	11.02	Air Monitoring
** Not Applicable means that an MF does not have a corresponding TER Section because it relates to the collection of information that NRC staff expects to be informative of future SDF performance, but was not relevant to TER conclusions.*		

Table 1-6 provides the NRC prioritization of each SDF MF under MAs 1 – 5.

Table 1-6: NRC Prioritization of Monitoring Factors under Monitoring Areas 1 - 5

MA 1 Inventory	MA 2 Infiltration and Erosion Control	MA 3 Waste Form Hydraulic Performance	MA 4 Waste Form Physical Degradation	MA 5 Waste Form Chemical Degradation
- 1.01 - Inventory in Disposal Structures *§*	- 2.01 - Hydraulic Performance of Closure Cap *†*	- 3.01 - Hydraulic Conductivity of Field-Emplaced Saltstone ±	- 4.01 - Waste Form Matrix Degradation ±	- 5.01 - Radionuclide Release from Field-Emplaced Saltstone ±
- 1.02 - Methods Used to Assess Inventory *‡*	- 2.02 - Erosion Protection *†*	- 3.02 - Variability of Field-Emplaced Saltstone ±	- 4.02 - Waste Form Macroscopic Fracturing ±	- 5.02 - Chemical Reduction of Tc by Saltstone ±
		- 3.03 - Applicability of Laboratory Data to Field-Emplaced Saltstone ±		- 5.03 - Reducing Capacity of Saltstone *‡*
		- 3.04 - Effect of Curing Temperature on Saltstone Hydraulic Properties ±		- 5.04 - Certain Risk-Significant K_d Values for Saltstone *‡*
				- 5.05 - Potential for Short-Term Rinse-Release from Saltstone *‡*
§ Periodic Monitoring Factors (i.e., MFs related to data that NRC staff expects to review on a periodic basis)				
† Low Priority				
‡ Medium Priority				
± High Priority				

Table 1-7 provides NRC prioritization of each SDF MF under MAs 6 – 9 and MA 11. Note that MA 10 is in Table 1-8.

Table 1-7: NRC Prioritization of Monitoring Factors under Monitoring Areas 6 – 9, and 11

MA 6 Disposal Structure Performance	MA 7 Subsurface Transport	MA 8 Environmental Monitoring	MA 9 Site Stability	MA 11 Radiation Protection Program
- 6.01 - Certain Risk-Significant K_d Values in Disposal Structure Concrete ‡	- 7.01 - Certain Risk-Significant K_d Values in Site Sand and Clay ‡	- 8.01 - Leak Detection §	- 9.01 - Settlement Due to Increased Overburden ‡	- 11.01 - Dose to Individuals During Operations §
- 6.02 - Tc Sorption in Disposal Structure Concrete ±		- 8.02 - Groundwater Monitoring §	- 9.02 - Settlement Due to Dissolution of Calcareous Sediment ‡	- 11.02 - Air Monitoring §
- 6.03 - Performance of Disposal Structure Roofs and HDPE/GCL Layers ‡				
- 6.04 - Disposal Structure Concrete Fracturing ‡				
- 6.05 - Integrity of Non-cementitious Materials ‡				
§ Periodic Monitoring Factors (i.e., MFs related to data that NRC staff expects to review on a periodic basis)				
† Low Priority				
‡ Medium Priority				
± High Priority				

Table 1-8 provides NRC staff prioritization for MA 10 (Performance Assessment Model Revisions). In the NRC 2012 TER, NRC staff concluded that Case K1 represented the most realistic case of those cases that DOE presented in the PA and RAI responses. So, NRC staff relied heavily on Case K1 in its review. The NRC prioritization for MA 10 is based on the risk-significance of the item in Case K1. If different assumptions are made in a subsequent PA, then the relative risk-significance of that item may change.

Table 1-8: NRC Prioritization of Monitoring Factors under Monitoring Area 10

Monitoring Factors	Additional Notes
- 10.01 - Implementation of Conceptual Models ±	
- 10.02 - Defensibility of Conceptual Models ±	
- 10.03 - Diffusivity in Degraded Saltstone ‡	
- 10.04 – K_d Values for Saltstone †	This could become more risk-significant in future assessments. For example, if assumed K_d values or inventories of certain radionuclides change, then the assumed K_d values could be more risk-significant than the K_d values used in Case K1. This MF does not include K_d values for radionuclides considered under MA 5, which have greater risk-significance in Case K1.
- 10.05 - Moisture Characteristic Curves †	In Case K1, MCCs were not used. If MCCs are used in the future, then this factor could be much more risk-significant.
- 10.06 - K_d Values for Disposal Structure Concrete †	This could become much more risk-significant in future assessments. For example, if assumed K_d values or inventories of certain radionuclides change, then the assumed K_d values could be more risk-significant than the K_d values used in Case K1. This MF does not include K_d values for radionuclides considered under MA 6, which have greater risk-significance in Case K1.
- 10.07 - Calculation of Build-Up in Biosphere Soil †	
- 10.08 - Consumption Factors and Uncertainty Distributions for Transfer Factors ‡	
- 10.09 - K_d Values for SRS Soil †	This could become much more risk-significant in future assessments. For example, if assumed K_d values or inventories of certain radionuclides change, then the assumed K_d values could be more risk-significant than the K_d values used in Case K1. This MF does not include the K_d values for radionuclides considered under MA 7, which have greater risk-significance in Case K1.
- 10.10 - Far-Field Model Calibration ‡	
- 10.11 - Far-Field Model Source Loading Approach ‡	
- 10.12 - Far-Field Model Dispersion ‡	
- 10.13 - Impact of Calcareous Zones on Contaminant Flow and Transport †	
§ Periodic Monitoring Factors (i.e., MFs related to data that NRC staff expects to review on a periodic basis)	
† Low Priority	
‡ Medium Priority	
± High Priority	

1.8.1. Relationship Between Previous Open Issues and Monitoring Factors

Since 2006, the NRC has been performing monitoring activities at the SDF. While monitoring under SDF Monitoring Plan, Rev. 0 *(NRC, 2007a),* NRC identified some Open Issues that had not been closed as of the NRC 2012 TER *(NRC, 2012)*. Therefore, in that TER, NRC identified concerns for future monitoring activities, including the concerns from the Open Issues that had not been closed previously. The MFs in this monitoring plan (i.e., Rev. 1) cover both the NRC concerns in the NRC 2012 TER and the previous Open Issues that had not been closed.

Table 1-9 describes where the concerns in those previous Open Issues (now administratively closed) are in the MFs under this monitoring plan.

Table 1-9: Link Between Previous Open Issues and Monitoring Factors

Previous Open Issue / Now, Administratively Closed	Included in Monitoring Factors
2007-1: DOE should determine the hydraulic and chemical properties of as-emplaced saltstone grout.	3.01, 3.02, 3.03, 3.04, and 5.01
2007-2: DOE should demonstrate that intrabatch variability, flush water additions to freshly poured saltstone grout at the end of each production run, and additives used to ensure processability are not adversely affecting the hydraulic and chemical properties of the final saltstone grout.	3.02 and 5.01
2009-1: DOE should demonstrate that: (i) Tc-99 in salt waste is strongly retained in saltstone grout, and (ii) the sorption of dissolved Tc-99 onto saltstone grout and disposal structure concrete is consistent with the K_d values for Tc-99 assumed in the 2009 PA.	5.01, 5.02, and 5.03

1.8.2. Closing Monitoring Factors

In general, an MF is closed by NRC when NRC determines that the specific technical concern associated with that MF is resolved. In addition, NRC closing an MF <u>does not</u> imply that DOE disposal actions met the PO related to that MF. At the end of each MF section in this monitoring plan, there is a specific paragraph that provides the current NRC expectations of how and/or when NRC expects that the MF will be closed. There is no way to “partly close” an MF, however that MF may be closed with a new MF created with the part of the old MF that was not closed.

Closing an MF is directly related to whether or not the NRC concluded in a TER or other document (e.g., onsite observation visit report) that DOE disposal actions were in compliance with the PO related to that MF. Although NRC staff may find in the future that other general conditions exist to close an MF, the following are general conditions that NRC expects to use to close an MF under this monitoring plan.

If the NRC did conclude that DOE demonstrated compliance with meeting a PO, then NRC expects to close an MF related to that PO if at least one of the following four conditions exists:

- NRC concludes that appropriate assumptions or parameters were consistent with the related assumptions in the PA.
- NRC concludes that appropriate conservative assumptions were made.
- NRC concludes that adequate support was provided for experimental values.

- NRC concludes that experimental values are well-supported and additional measurements would be unlikely to result in a different estimate (i.e., NRC would be satisfied that additional DOE work on the MF would be unnecessary).

If NRC did not conclude that DOE demonstrated compliance with meeting a PO, then NRC expects to close an MF related to that PO if at least one of the following two conditions exists:

- NRC concludes that appropriate conservative assumptions were made (i.e., NRC expects that this would require both a new DOE PA and a new NRC TER).

- NRC concludes that experimental values are well-supported and additional measurements would be unlikely to result in a different estimate (i.e., NRC would be satisfied that additional DOE work on the MF would be unnecessary).

(Intentionally Left Blank)

2. MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.40

§61.40, "General requirement"

> *Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in §§61.41 through 61.44.*

The terms "land disposal facility," "reasonable," "sited," "designed," "operated," "closed," and "controlled after closure" are defined in the Definitions section of this monitoring plan. NRC does not define in its regulations or guidance the term, "reasonable assurance."

§61.40 is a general requirement in the regulation rather than a specific requirement. Therefore, if the NRC concludes with reasonable assurance that DOE will meet the other four POs (i.e., §61.41 through §61.44), then the NRC concludes with reasonable assurance that DOE will meet §61.40. Otherwise, the NRC cannot conclude with reasonable assurance that DOE will meet §61.40. Therefore, there are no specific MAs or MFs for §61.40 in this monitoring plan, and §61.40 will not be mentioned further in this monitoring plan.

(Intentionally Left Blank)

3. <u>MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.41</u>

§61.41, "Protection of the general population from releases of radioactivity"

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

The term "as low as is reasonably achievable" is defined in the Definitions section of this monitoring plan.

As described in NUREG-1854 *(NRC, 2007b)*, the NRC will evaluate DOE compliance with the §61.41 PO by using a dose limit of 0.25 mSv/yr (25 mrem/yr) Total Effective Dose Equivalent (TEDE). That approach is consistent with the approach described in the final rule for 10 CFR Part 63 *(66 FR 55752)*:

> *Because each of the organs had the same limit under the older system even though each had a different level of radiosensitivity, it is very difficult to directly compare the old standards with the new standards. As noted in the proposed rule, the Commission considers 0.25 mSv/yr (25 mrem/yr) TEDE as the appropriate dose limit to compare with the range of potential doses represented by the older limits that had whole body dose limits of 0.25 mSv/yr (25 mrem/yr).*

The DOE 2009 PA and the NRC 2012 TER *(NRC, 2012)* used the most up-to-date dose factors to calculate the potential dose, which were consistent with Commission direction in SRM-SECY-01-0148, "Processes for Revision of 10 CFR Part 20 Regarding Adoption of ICRP Recommendations on Occupational Dose Limits and Dosimetric Models and Parameters" *(NRC, 2001)*. The dose assessment in the DOE 2009 PA for the member of the public evaluated well water, surface water, and air pathways. Groundwater was assumed to be withdrawn from a well at the point of maximum exposure 100 meters from the SDF. In the NRC 2012 TER, NRC concluded that the scenarios and pathways considered by DOE were appropriate based on the regional practices near SRS.

The objective of the NRC monitoring activities related to §61.41 is for NRC to assess whether DOE disposal of salt waste at the SDF meets the PO. NRC monitoring of salt waste disposal at the SDF for compliance with §61.41 differs from NRC monitoring activities for the other three POs because in the NRC 2012 TER, NRC concluded it did not have reasonable assurance that DOE disposal actions would comply with §61.41. In the NRC 2012 TER, the NRC concluded it had reasonable assurance that DOE disposal actions would comply with §61.42 – §61.44. The practical implication of that difference is that future DOE demonstrations of consistency with the DOE 2009 PA assumptions or parameters is not a valid basis for NRC closing the §61.41 MFs.

Generally, an MF may be closed when adequate support is provided for an experimental value. However, a well-supported value may or may not support an NRC conclusion with reasonable assurance of DOE compliance with §61.41. Generally, an MF may be closed if NRC determines that additional experimental work is unlikely to yield a different value. For example, if DOE conducts a series of measurements that yield a consistent value for the hydraulic conductivity of field-emplaced saltstone, then, whether or not this value is consistent with meeting §61.41, that MF may be closed if NRC determines that the value is well-supported and additional measurements are unlikely to result in a different estimate. That is because the NRC would be satisfied that additional DOE work on that MF is unnecessary. A determination that an MF is closed does not necessarily mean that NRC can conclude with reasonable assurance that §61.41 will be met by DOE.

Generally, an MF may be closed if the NRC determines that a conservative assumption can be made that is consistent with meeting §61.41. In some cases, additional DOE experimental work could result in refined assumptions or parameter values and DOE could prefer to make those defensible conservative assumptions in parameter values. However, because NRC concluded in the 2012 TER that it did not have reasonable assurance that §61.41 would be met, NRC expects that using conservative assumptions in parameter values would not show that §61.41 is met without other significant changes to the PA.

Table 1-2 contains the listing of the MAs and their associated POs. That table shows that the MAs related to §61.41 are MA 1 through MA 10: Inventory; Infiltration and Erosion Control; Waste Form Hydraulic Performance; Waste Form Physical Degradation; Waste Form Chemical Degradation; Disposal Structure Performance; Subsurface Transport; Environmental Monitoring; Site Stability; and Performance Assessment Model Revisions.

Inventory (MA 1) is important to §61.41 because DOE assumed radionuclide release is limited by sorption and DOE does not take credit for potential limits on radionuclide solubility, which means that Inventory is linearly related to potential dose. Infiltration and Erosion Control (MA 2) is important to §61.41 because the closure cap is designed to limit Infiltration and provide Erosion Control. Waste Form Hydraulic Performance (MA 3) is important to §61.41 because the effects of the hydraulic properties of saltstone on the rate of radionuclide release into groundwater are important to SDF performance. Waste Form Physical Degradation (MA 4) is important to §61.41 and is critical to SDF performance because the physical integrity of saltstone plays an important role in both limiting water infiltration and limiting saltstone oxidation, which has a significant effect on Tc-99 release.

Waste Form Chemical Performance (MA 5), Disposal Structure Performance (MA 6), and Subsurface Transport (MA 7) are important to §61.41 and SDF performance primarily because of the critical importance of radionuclide sorption in controlling the magnitude and the timing of the peak dose. In general, more sorption in the source material limits the peak dose by lowering the radionuclide's annual fractional release rate (i.e., higher sorption coefficient in the source reduces the quantity of a radionuclide released in any one year). Sorption in downstream areas, such as the disposal structure or vadose zone, delays radionuclide transport to the receptor site, which can reduce the peak dose from short-lived radionuclides and result in attenuation of pulse-like releases. Also, in DOE Case K, Case K1, and Case K2 analyses, DOE modeled significantly more Tc-99 sorption in disposal structure concrete than in saltstone, resulting in re-concentration of Tc-99 in the disposal structure. DOE modeled Tc-99 concentration in disposal structure concrete became greater than the original modeled Tc-99 concentration in saltstone. Thus, disposal structure concrete became a *de facto* new source. Therefore, increasing K_d values in disposal structure concrete directly limits the modeled peak dose in the DOE Case K model in SRR-CWDA-2011-00044 by limiting the annual fractional release of Tc-99 into the environment.

Environmental Monitoring (MA 8) is important to §61.41 because the radionuclide and relevant chemical concentrations in a variety of environmental media (e.g., soil, air, surface water, animal meat, groundwater) are important. Of the DOE monitoring measurements for those radionuclides and concentrations, NRC staff expects groundwater monitoring to be most important to the demonstration of compliance with §61.41 because the potential dose from the SDF to an off-site receptor will be dominated by groundwater pathways. Groundwater monitoring also includes measurements of other parameters that may serve as early indicators of releases from the SDF (e.g., groundwater nitrate concentrations, pH). Environmental Monitoring includes monitoring of data from the leak detection system that DOE has informed NRC has been installed under SDS 3A and could be installed under other disposal structures.

Site Stability (MA 9) is important to NRC assessing DOE compliance with §61.41, §61.42, and §61.44. Site Stability is important in limiting the infiltration through the SDF, which is important in maintaining compliance with both §61.41 and §61.42. Site Stability also is important to maintaining an adequate barrier to intrusion, which is important in maintaining compliance with §61.42. By definition, Site Stability is important to NRC assessing DOE compliance with §61.44. Performance Assessment Model Revisions (MA 10) is important to §61.41 and §61.42 because the MFs under MA 10 are items that are cross-cutting among the other MAs and NRC staff does not expect them to be closed before DOE updates the PA. Thus, they are items that NRC staff will review when DOE next revises the PA under the DOE PA maintenance program.

3.1. MA 1 – Inventory

The dose projected in the DOE 2005 and 2009 PAs is directly related to the assumed inventory. Thus, the inventory disposed of at the SDF was identified as an MF during the NRC 2005 review and documented in the NRC 2005 TER *(NRC, 2005)*. Since 2006, the NRC has been monitoring the inventory disposed at the SDF. The methodology used to quantify and track the inventory of radionuclides in liquid waste that is transferred to the SDF was discussed and reviewed by NRC staff during five onsite observation visits (i.e., October 2007, March 2008, March 2009, February 2010, April 2011) and the NRC conclusions were documented in the Observation Reports *(NRC, 2008a; NRC, 2008b; NRC, 2009b; NRC, 2010a; NRC, 2011b)*.

As waste continues to be disposed at the SDF, the NRC will continue to monitor the inventory on a radionuclide-specific basis in each disposal structure. Also, NRC staff will continue to review the methodology used by DOE to quantify the inventory to confirm that risk-significant radionuclides are being assessed adequately. Although a risk-informed NRC review is expected to focus on the radionuclides that were identified as potentially risk-significant (i.e., Tc-99, I-129, Ra-226, Se-79, the ancestors of Ra-226), NRC staff will review the inventory of the other radionuclides to confirm that they are not present at risk-significant levels.

3.1.1. MF 1.01: Inventory in Disposal Structures

The inventory in an individual disposal structure is relevant to the protection of the general public because a small number of disposal structures can dominate the dose to an off-site member of the public. Although certain disposal structures dominate the dose projected in the DOE 2009 PA, the location of the peak dose and degree of plume overlap can change with changing assumptions about far-field transport. The relative inventory in different disposal structures can result in certain disposal structures becoming more dose-significant. The NRC will monitor the inventory disposed of in all disposal structures rather than just those disposal structures that contributed most significantly to the dose calculated in the DOE 2009 PA.

The NRC will monitor the inventory in each disposal structure in comparison to the values in Table A-1 (Projected Inventory at Time of Closure) of this monitoring plan. The inventory values listed in Table A-1 are from the DOE 2009 PA for all radionuclides, except for Ra-226, Th-230, U-234, and Pu-238. The inventory for those four radionuclides was based on the revised inventory provided by DOE to NRC staff in 2012 (SRR-CWDA-2011-0004). If the inventory of a radionuclide in a particular disposal structure is higher than the value in Table A-1, then NRC staff expects that DOE would perform an analysis to understand the effect of the increased inventory on the dose. For radionuclides that are not significant contributors to the total dose, the dose consequence may be minimal. However, a large increase in the inventory of a radionuclide that was not projected to be a significant contributor to the dose in the DOE 2009 PA may result in that radionuclide becoming risk-significant. The analysis should include the increase in dose from that disposal structure and consider the dose from neighboring disposal structures that are based on the final inventory of those disposal structures.

NRC expects to close MF 1.01 (Inventory in Disposal Structures) under PO §61.41 after DOE has completed waste disposal at the SDF and determined the final inventory in each disposal structure.

3.1.2. MF 1.02: Methods used to Assess Inventory

NRC will monitor the methods DOE uses to assess radionuclide inventories because of the considerable uncertainty in inventory estimates and the importance of radionuclide inventory to dose to an off-site member of the public. NRC will monitor the approach for sampling the waste, the measurement of radionuclides in the waste samples, the methods used to estimate the concentration of radionuclides present at levels below the detection limit, and the methods used to track the inventory in disposal structures. NRC will focus on radionuclides that are currently identified as risk-significant and their relevant ancestors (e.g., Tc-99, Ra-226, Th-230, I-129), but will consider radionuclides that could become more risk-significant if the inventory increases significantly or if the DOE modeling assumptions change.

There is significant uncertainty in the inventory of Ra-226 and its parent, Th-230, because those radionuclides are present in the salt waste at levels that are below the detection limit of the analytical method used by DOE. NRC staff scoping calculations in the NRC 2012 TER (*NRC, 2012)* based on the DOE Case K indicate that the Ra-226 dose is potentially significant if those radionuclides are present at levels that are as high as the detection limit. That is especially important for tanks that are known to contain Th-230 waste. If the results from any experiments that DOE performs to address the other MFs related to the potential dose from Ra-226 (i.e., MFs 5.04 and 6.01) demonstrate that the potential Ra-226 dose is acceptable, even if Ra-226 and Th-230 are present at the detection limits, then this aspect of MF 1.02 will no longer be relevant.

The methodology used to quantify and track the inventory of radionuclides in liquid waste that is transferred to the SDF was discussed and reviewed by NRC staff during four onsite observation visits (i.e., October 2007, March 2008, February 2010, April 2011). Also, during the March 2008 visit, NRC staff reviewed the procedures DOE used for obtaining samples and the analytical procedures used for quantifying the constituents in the samples. In those Observation Reports *(NRC, 2008a; NRC, 2008b; NRC, 2010a; NRC, 2011b)*, NRC concluded that the methodology used to assess the inventory disposed of at the SDF appeared to be appropriate. However, because Ra-226 was not identified as a potentially significant dose contributor in the document (CBU-PIT-2005-00146), the quantification of Ra-226 and its ancestors was not reviewed in detail at that time. Additionally, since those visits, DOE made changes to the process used to determine the inventory (SRR-CWDA-2012-00002). Thus, NRC will continue to monitor the methods used to assess the inventory.

NRC expects to close MF 1.02 (Methods Used to Assess Inventory) under PO §61.41 after DOE has completed waste disposal at the SDF and determined the final inventory in each disposal structure.

3.2. MA 2 – Infiltration and Erosion Control

The performance of the closure cap affects the overall performance of the SDF because the cap is designed to: (1) provide physical stabilization, (2) limit infiltration, and (3) act as an intruder deterrent. In the DOE 2009 PA, DOE projected that the sensitivity of the SDF performance to the cap would be limited. That projection was largely due to the DOE modeled shedding of water around the disposal structures by engineered layers above the individual disposal structures (i.e., not included as part of the cap). If the diversion of water around the disposal structures is determined to be optimistic, then the performance of the cap would become more risk-significant. Thus, NRC staff is interested in information relevant to the assumed cap performance.

The SDF cap design consists of two distinct closure parts with one part on the northwest side of the site and the other part on the southeast side of the site. Those two distinct parts are considered as components of one "closure cap" because they have the same design. From the surface downward, the cap consists of topsoil and backfill, an erosion barrier of coarse stone, a middle backfill layer, an upper drainage layer over a high-density polyethylene (HDPE)/Geosynthetic clay liner (GCL) composite layer, and additional backfill (see Figure 3-1).

Also, several geotextile filter fabric layers in the cover, while not directly influencing flow, serve to separate some of the engineered layers. Figure 3-1 includes engineered layers designed to divert water around the individual disposal structures, which are not considered part of the cap. Those layers are located immediately above the disposal structure roofs and are discussed further in MF 6.03. Figure 3-1, which is adapted from a conceptual illustration of the components of the SDF cap and additional engineered layers above the disposal structures, is Figure 3.2-20 from the DOE 2009 PA.

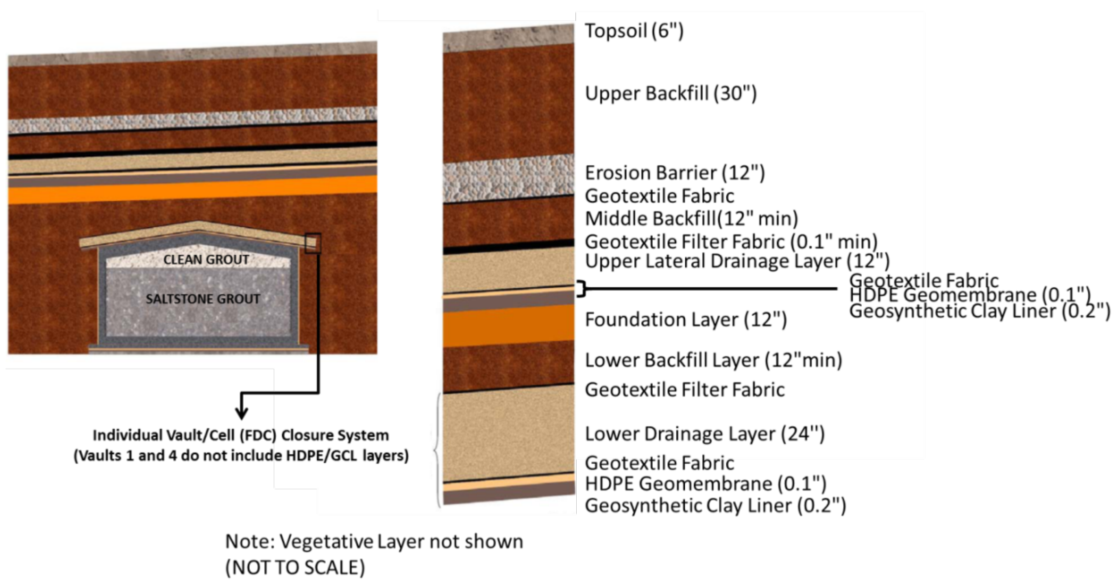

Figure 3-1: Conceptual Illustration of the Components of SDF Closure Cap and Additional Engineered Layers above the Disposal Structures

In the DOE 2009 PA, the performance of the cap was based on the preliminary DOE design of the cap. NRC will monitor the development of specific designs for the cap and will determine if those designs are likely to significantly alter the projected cap performance in the DOE 2009 PA. Prior to any construction activities, NRC staff should review the specifications for cover construction materials and the quality assurance/quality control procedures for ensuring that those materials meet the specifications. During construction, NRC staff should observe the placement of those materials and the quality control testing to determine whether or not the as-built cover meets the design specifications.

3.2.1. MF 2.01: Hydraulic Performance of Closure Cap

In the DOE 2009 PA, the closure cap was assumed to limit the amount of water interacting with the saltstone waste form. That hydraulic performance is largely dependent on: (1) the performance of the HDPE/GCL composite layer, (2) the hydraulic conductivity of the lower foundation layer, and (3) the evapotranspiration from the vegetative cover and topsoil layers. In the NRC 2012 TER *(NRC, 2012)*, NRC staff described the need for additional model support for the assumed long-term performance of the cap.

Based on the current cap design and model results, NRC staff should focus on the assumed performance of the HDPE/GCL composite layer and the lower foundation layer. Model support for the cap may include pilot-scale testing of the cap design; laboratory results and literature reviews related to the performance of cap components; studies of analog sites to estimate long-term infiltration; and revised simulations of cap performance.

NRC expects to close MF 2.01 (Hydraulic Performance of Closure Cap) under PO §61.41 after NRC determines that the hydraulic performance of the as-built closure cap is adequate. Given the importance of construction activities on the performance of the cap, MF 2.01 will not be closed prior to construction of the cap.

3.2.2. MF 2.02: Erosion Protection

The ability of the closure cap to resist erosion is important in maintaining the barriers to infiltration within the cap. The erosion protection design is important in maintaining a minimum of 3 meters (10 feet) of clean material above the disposal structures to deter inadvertent intrusion. NRC will monitor the DOE erosion protection designs (i.e., design changes, implementation) to verify that those assumptions related to cap performance are met. NRC will monitor information related to the physical stability of the vegetative and topsoil layers. NRC staff should evaluate the ability of those cover layers to withstand the effects of high frequency/low intensity rainfall events, which can dominate long-term erosion. NRC staff should evaluate the projected impacts of degradation (e.g., due to fire or drought) on the stability of the vegetative cover. NRC staff has not identified monitoring activities related to evapotranspiration from the vegetative cover and topsoil, other than expecting those layers are present and resistant to erosion.

NRC expects to close MF 2.02 (Erosion Protection) under PO §61.41 after NRC determines that the physical stability of the final closure cap is adequate. Given the importance of construction activities on the performance of the cap, MF 2.02 will not be closed prior to construction of the cap.

3.3. MA 3 – Waste Form Hydraulic Performance

The hydraulic properties of saltstone are important to SDF performance because of their effects on the rate of radionuclide release into the groundwater. NRC staff is primarily concerned with saltstone saturated hydraulic conductivity and diffusivity. Hydraulic conductivity is important because water flow through saltstone is limited in part by saltstone hydraulic conductivity. Water flow through saltstone directly affects radionuclide release into groundwater. Diffusivity is important for two reasons: (1) depending on the nature of long-term saltstone degradation, radionuclide release may be limited by diffusion of radionuclides from the saltstone matrix; and (2) diffusivity affects the rate of saltstone oxidation, which directly affects the rate of Tc release.

MA 3 (Waste Form Hydraulic Performance) primarily addresses the saltstone saturated hydraulic conductivity and diffusivity. In Case K, DOE assumed that the relative permeability of saltstone is 1 (i.e., hydraulic conductivity of saltstone does not decrease with decreasing saltstone saturation). If DOE uses Moisture Characteristic Curves (MCCs) to represent decreasing saltstone hydraulic conductivity with decreasing saturation in future analyses, then NRC will monitor activities related to those MCCs.

Previously, issues related to representativeness of laboratory-made samples and the variability of field-emplaced saltstone properties were tracked as Open Issues 2007-1 and 2007-2 because of the importance of saltstone hydraulic characteristics to SDF performance. Section 1.8.1 of this monitoring plan describes how those two previously Open Issues are now administratively closed with NRC staff concerns being under MFs in this monitoring plan: Open Issue 2007-1 is under MFs 3.01 through 3.04 and 5.01; and Open Issue 2007-2 is under MFs 3.02 and 5.01.

3.3.1. MF 3.01: Hydraulic Conductivity of Field-Emplaced Saltstone

The value of saltstone hydraulic conductivity DOE used in the Case A analysis was based on measurements of laboratory-made samples. The representativeness of those laboratory samples is addressed as MF 3.03. DOE previously collected core samples from SDS 4, Cell E (SRNL-STI-2009-00804, SRNL-STI-2010-00657), but DOE concluded that the results were not representative because of artifacts caused by the sampling method. To address that concern, DOE developed a formed-core technique to sample field-emplaced saltstone. DOE discussed that technique with NRC staff during an onsite observation visit in July 2010*(NRC, 2010c)*. NRC will monitor the values of hydraulic conductivity of field-emplaced saltstone samples collected with the formed-core technique. NRC will monitor hydraulic conductivity values from any other representative studies. NRC staff should evaluate the appropriateness of any new sampling technique. NRC staff expects that the model support will include measurements of the hydraulic conductivity of samples of field-emplaced saltstone. NRC staff should evaluate the applicability of the sample measurements to the whole saltstone wasteform, including any effects of differences in scale or boundary conditions.

NRC expects to close MF 3.01 (Hydraulic Conductivity of Field-Emplaced Saltstone) under PO §61.41 after NRC determines that model support for the saturated hydraulic conductivity of field-emplaced saltstone is sufficient.

3.3.2. MF 3.02: Variability of Field-Emplaced Saltstone

MF 3.02 focuses on the variability in the saltstone product. Variations in the composition and curing conditions of saltstone grout produced at the Saltstone Processing Facility (SPF) and emplaced in the disposal structures may affect hydraulic properties of saltstone grout. Quality assurance for the emplaced saltstone waste form is important for SDF performance because of the sensitivity of dose projections to saltstone hydraulic properties. Saltstone quality assurance was discussed during three onsite observation visits (i.e., October 2007, March 2008, July 2010), as documented in those reports *(NRC, 2008a; NRC, 2008b; NRC, 2010c)*. NRC will continue to monitor the potential variability of field-emplaced saltstone properties and the DOE quality assessment program for grout.

Variability may be measured directly with field samples, or indirectly with laboratory measurements. DOE conducted several experiments to determine the effects of variations in saltstone composition and curing conditions on saltstone hydraulic characteristics. To assess potential variability in field-emplaced saltstone, NRC staff should: (i) evaluate results of laboratory experiments in which saltstone composition and curing conditions are varied in ranges that might be experienced by field-emplaced saltstone; and (ii) evaluate whether or not the parameters related to the composition, placement, and curing of saltstone are limited to the ranges in which the hydraulic properties of the resulting waste form are well supported by DOE.

NRC expects to close MF 3.02 (Variability of Field-Emplaced Saltstone) under PO §61.41 after NRC determines that saltstone production, placement, and curing conditions that significantly affect saltstone hydraulic properties are well controlled.

3.3.3. MF 3.03: Applicability of Laboratory Data to Field-Emplaced Saltstone

MF 3.03 is closely related to MF 3.02, but focuses on the potential differences between laboratory and field conditions of field-emplaced saltstone. Specifically, it is unclear to NRC staff whether or not laboratory samples reflect the composition (e.g., actual water-to-cement ratios, admixtures, variations in aluminate concentrations in treated salt waste) and curing conditions (e.g., actual curing temperature, humidity) of field-emplaced saltstone. Also, it is unclear to NRC staff whether or not laboratory samples adequately capture the effects of scale (e.g., effects of fractures or interfaces between saltstone lifts).

DOE performed several experiments to determine the effects of variations in saltstone composition and curing conditions on saltstone hydraulic properties. NRC staff will continue to evaluate the results of those and similar studies to determine if laboratory samples used as a basis for the hydraulic properties of saltstone used in Case A of the DOE 2009 PA adequately represent field conditions (e.g., oxygen partial pressure, saltstone moisture content) DOE expects to evolve once the closure cap is in place and as the performance period continues. NRC staff should evaluate comparisons of the properties of laboratory-produced and field-emplaced samples to determine the extent to which laboratory-produced samples can be relied upon as a basis for values used in the DOE 2009 PA.

NRC expects to close MF 3.03 (Applicability of Laboratory Data to Field-Emplaced Saltstone) under PO §61.41 after NRC determines that representing the hydraulic properties of field-emplaced saltstone with the hydraulic properties of laboratory-produced samples is adequate. That assessment should account for the range of expected disposal conditions of field-emplaced saltstone as well as effects of scale. Alternately, MF 3.03 may be closed if NRC determines that DOE bases the hydraulic properties of saltstone on the properties of an appropriate range of samples of field-emplaced saltstone, rather than on measurements of laboratory-produced samples.

3.3.4. MF 3.04: Effect of Curing Temperature on Saltstone Hydraulic Properties

DOE performed several experiments to demonstrate the importance of the curing temperature profile (i.e., curing temperature as a function of time) to saltstone hydraulic properties. However, DOE indicated that the results of some of those experiments may not be reliable because humidity was not adequately controlled. NRC will continue to monitor the development of information about the effect of curing temperature profiles on saltstone hydraulic properties. NRC will continue to monitor data from thermocouples embedded in field-emplaced saltstone. NRC staff should compare information about the effect of curing temperature profile on saltstone hydraulic properties to the measured curing temperature profiles of field-emplaced saltstone.

NRC expects to close MF 3.04 (Effect of Curing Temperatures on Saltstone Hydraulic Properties) under PO §61.41 after NRC determines that projected SDF performance is based on estimates of the hydraulic properties of saltstone (e.g., hydraulic conductivity and diffusivity) that are well-supported. That support should account for the range of curing conditions (i.e., temperatures values, humidity values) experienced by field-emplaced saltstone.

3.4. MA 4 – Waste Form Physical Degradation

The physical degradation of saltstone is important in large part due to the effects of degradation on the saltstone hydraulic properties. The NRC staff expects that physical degradation will lead to increased water flow through both the saltstone matrix and macroscopic fractures, which will increase the projected dose. The increased water flow can increase the rate of subsequent saltstone degradation. The NRC staff expects that physical degradation will increase both radionuclide diffusion from saltstone and oxygen diffusion into saltstone. NRC staff expects that faster radionuclide diffusion out of saltstone will affect dose directly by increasing the annual fractional release rate of radionuclides into groundwater. NRC staff expects that faster oxygen diffusion into saltstone will increase the dose from redox sensitive radionuclides, many of which are more mobile when oxidized.

The distinction between matrix degradation and macroscopic fracturing is important largely because different experiments and measurement methods may be used to quantify the “bulk” hydraulic properties of the matrix, as compared to the formation of macroscopic fractures. Also, matrix degradation and macroscopic fracture formation are often represented separately in the DOE PA model.

MA 4 (Waste Form Physical Degradation) is closely related to MA 3 (Waste Form Hydraulic Performance) because matrix degradation is expected to be quantified primarily through its effects on saltstone hydraulic conductivity and diffusivity. The main difference between MA 4 and MA 3 is that MA 4 focuses on the changes in matrix properties and macroscopic fractures with time; while MA 3 focuses primarily on the initial properties.

3.4.1. MF 4.01: Waste Form Matrix Degradation

In Case A, DOE projected that the physical properties of saltstone would not change during the performance period. In the NRC 2012 TER *(NRC, 2012)*, NRC staff determined that assumption was not realistic. In the DOE response to NRC staff concern, DOE indicated that sensitivity analyses capture the potential effects of saltstone degradation. However, in the NRC 2012 TER, the NRC concluded that none of the DOE sensitivity analyses provided reasonable assurance that the DOE planned disposal actions met the PO in §61.41.

The SDF performance is sensitive to the rate of water flow through saltstone. Also, increased saltstone diffusivity is expected to increase the release of redox sensitive radionuclides by increasing saltstone oxidation. SDF performance is also sensitive to the rate of increase of saltstone hydraulic conductivity and diffusivity because sudden degradation of saltstone can result in pulse-like releases that would cause a higher annual dose than a more gradual release. Sudden hydraulic degradation may be particularly detrimental if it occurs many thousands of years after site closure, after significant oxidation has occurred. NRC will monitor the development of DOE model support for projected changes in waste form hydraulic conductivity and diffusivity during the performance period.

This model support may include the results of mechanistic modeling and the results of laboratory experiments intended to simulate accelerated aging. NRC staff should begin by becoming familiar with ongoing and planned DOE research on saltstone degradation as described in the DOE response to the NRC second RAI. NRC staff should coordinate this review with the reviews for MFs 3.01 through 3.04 because the MFs are related.

NRC expects to close MF 4.01 (Waste Form Matrix Degradation) under PO §61.41 after NRC determines that support for modeled changes in the saturated hydraulic conductivity and diffusivity during the performance period is sufficient.

3.4.2. MF 4.02: Waste Form Macroscopic Fracturing

In Case A, DOE projected that saltstone would not fracture during the performance period. In the NRC 2012 TER *(NRC, 2012)*, NRC concluded that it was unrealistic to assume that saltstone will remain intact for thousands of years after site closure. Also, that assumption was inconsistent with observed fractures in emplaced saltstone. In Case K, DOE assumed saltstone will fracture significantly during the performance period. However, in that TER, NRC concluded that none of the DOE sensitivity analyses, including Case K, provided reasonable assurance that the DOE planned disposal activities met the PO in §61.41.

Saltstone fracturing is important to site performance because it increases water flow through the saltstone, shortens diffusive pathways for radionuclide release, and provides additional surface area for the progression of saltstone oxidation, which increases Tc release. In the NRC 2012 TER, NRC staff described sensitivity analyses that indicated that SDF performance is sensitive to both the assumed degree of fracturing during the performance period and the rate of fracture formation.

The NRC will monitor the development of model support for the assumed degree of saltstone fracturing and the rate of saltstone fracture formation within the performance period. Although fracturing may be represented non-mechanistically, the NRC staff should evaluate whether or not the assumed fracture rate is likely to adequately represent saltstone fracturing during the performance period. If the uncertainty is large, then it would be appropriate to evaluate a range of potential fracture rates. The NRC staff should closely monitor the DOE model support for long-term saltstone fracturing, including research addressing the technical concerns. This model support may include the results of formal expert elicitation (e.g., NUREG-1563, “Branch Technical Position on the Use of Expert Elicitation in the High-Level Waste Program”) because of the lack of physical evidence to support projections of saltstone fracturing for thousands of years after emplacement. This model support may include mechanistic modeling and the results of laboratory experiments intended to simulate accelerated aging.

NRC expects to close MF 4.02 (Waste Form Macroscopic Fracturing) under PO §61.41 after NRC determines that model support for the assumed formation of macroscopic fractures during the performance period is sufficient.

3.5. MA 5 – Waste Form Chemical Degradation

The saltstone waste form is a cementitious material made by mixing treated salt waste solutions with a dry mixture of blast furnace slag, fly ash, and cement. Chemically, saltstone is initially alkaline and reducing. Those chemical properties of saltstone are expected to change over time as infiltrating groundwater depletes the reductive and buffering capacity of the saltstone matrix. That will cause the E_h to rise (i.e., it becomes less reducing) and the pH to fall. The initial chemical conditions and the evolution of those conditions over time can strongly affect the ability of the waste form to retain radionuclides. That will ultimately influence the release of contaminants from the waste form into the environment. For example, the release of redox sensitive elements, most notably Tc, is strongly influenced by E_h conditions. Tc is relatively insoluble and is immobile under reducing conditions, but is mobile under oxidized conditions.

In the DOE 2009 PA, the release of radionuclides from the saltstone waste form was modeled using a sorption coefficient (i.e., K_d value). In the NRC 2005 TER *(NRC, 2005)* and the subsequent monitoring activities, NRC staff identified the importance of the waste form chemical performance to the expected dose. Open Issues 2007-1 and 2007-2 related, in part, to the chemical properties of the field-emplaced saltstone and the effect of variability in the saltstone on the chemical properties. Open Issue 2009-1 related to the reduction and retention of Tc in saltstone. Section 1.8.1 of this monitoring plan describes how those three previously Open Issues are now administratively closed with NRC staff concerns being addressed under the MFs in this monitoring plan: Open Issue 2007-1 is under MFs 3.01 through 3.04 and 5.01; Open Issue 2007-2 is under MFs 3.02 and 5.01; and Open Issue 2009-1 is under MFs 5.01 through MF 5.03.

NRC staff will focus on radionuclides that are potentially significant to the dose. The NRC will monitor the release of radionuclides from field-emplaced saltstone, the chemical reduction of Tc by saltstone, the reducing capacity of saltstone, the sorption in or leaching of risk-significant radionuclides from saltstone, and the potential for short term rinse release from saltstone (i.e., a mechanism by which radionuclides are rinsed off of the surface of a waste form).

3.5.1. MF 5.01: Radionuclide Release from Field-Emplaced Saltstone

It is important to measure the release rates of radionuclides from field-emplaced saltstone because of the importance of radionuclide release rates to the projected dose. MF 5.01 is closely related to MFs 5.02 and 5.04. MF 5.01 focuses on measuring the release rates from field-emplaced saltstone. MFs 5.02 and 5.04 focus on the expected sorption coefficients or release rates from saltstone, which may be addressed using lab simulated saltstone.

Most of the DOE leaching experiments conducted reflected the bulk constituents of saltstone and simulated waste, but have not included the full suite of admixtures used in the production of saltstone at the SPF. It will be important for DOE to determine the rate of radionuclide leaching from samples containing those admixtures because certain admixtures, such as the anti-foam agent Tributyl Phosphate (TBP), may form chemical complexes with radionuclides that limit radionuclide sorption or increase solubility.

NRC expects to close MF 5.01 (Radionuclide Release from Field-Emplaced Saltstone) under PO §61.41 after NRC determines that measurements of radionuclide release rates from field-emplaced saltstone used in the PA are reliable.

3.5.2. MF 5.02: Chemical Reduction of Tc by Saltstone

Studies that DOE has relied on to demonstrate Tc retention in saltstone included experimental artifacts (e.g., presence of sodium thiosulfate or $H_2(g)$), which made it difficult to interpret the results. Based on those experiments, it is unclear to NRC staff whether or not saltstone itself can reduce Tc and maintain Tc in a reduced state. Furthermore, recent DOE studies showed the sensitivity of Tc retention in saltstone to trace quantities of oxygen (e.g., 30 ppm to 60 ppm). The peak-dose to an off-site member of the public and the inadvertent intruder are both sensitive to Tc release, which is sensitive to the Tc redox state. Thus, NRC needs a robust demonstration of the ability of saltstone to reduce Tc and maintain Tc in a reduced state in environmental conditions similar to the expected environmental conditions of the emplaced waste form. MF 5.02 is closely related to MF 5.01 and any information gathered under MF 5.01 may be useful in addressing MF 5.02.

Based on current NRC assessments, NRC needs to determine that DOE has robust model support for the chemical reduction of Tc(VII) to Tc(IV). NRC staff should determine whether the K_d values for Tc in saltstone, initially and as the waste form oxidizes, are appropriate and supported by experimental evidence.

NRC expects to close MF 5.02 (Chemical Reduction of Tc by Saltstone) under PO §61.41 after NRC determines that: (1) model support for the chemical reduction of Tc(VII) to Tc(IV) is robust; and (2) this reduced state is maintained under field conditions. NRC expects that DOE will inform NRC what the ranges of those conditions are expected to be during the performance period.

3.5.3. MF 5.03: Reducing Capacity of Saltstone

The DOE Case K model in the DOE 2009 PA and NRC sensitivity analyses in the NRC 2012 TER *(NRC, 2012)* demonstrated the importance of saltstone reducing capacity to the projected Tc release rate. However, it is unclear to NRC staff why the measured value of the specific reducing capacity of saltstone, which contains only 25% blast furnace slag by weight (i.e., slag represents 45% of the dry pre-mix by weight and represents 25% of the mass of finished saltstone, which is composed of dry pre-mix and salt waste.), is equivalent to the measured reducing capacity of pure blast furnace slag. Also, there is uncertainty in the E_h transition times assumed in the DOE 2009 PA, which affects the projected release of redox sensitive radionuclides other than Tc (e.g., Se) because Tc was not modeled with the pore-volume step-change release model used for all of the other elements. NRC will monitor the development of additional information regarding the initial reducing capacity of saltstone and the expected evolution of redox conditions over time.

NRC expects to close MF 5.03 (Reducing Capacity of Saltstone) under PO §61.41 after NRC determines that information for the initial reducing capacity of saltstone and the expected evolution of redox conditions over time is adequate.

3.5.4. MF 5.04: Certain Risk-Significant K_d Values for Saltstone

NRC will monitor measurements of Ra and Se sorption in or leaching from saltstone, which was represented using a K_d value in the DOE 2009 PA. In the NRC 2012 TER *(NRC, 2012)*, NRC determined that there was not an adequate basis for the K_d values DOE used to represent Ra and Se sorption in saltstone. MF 5.04 is currently limited to Ra and Se, but may be expanded to include other radionuclides if the K_d values in saltstone for the other radionuclides prove to be risk-significant and are not adequately supported by DOE.

MF 5.04 is closely related to MF 5.01 and MF 5.02. The K_d value for Tc in saltstone is under MF 5.02 because of its significant effect on the projected dose. The sorption coefficients for radionuclides that have the potential to affect the dose, but are less risk-significant are under MF 10.04. Information gathered under MF 5.01 may be useful in addressing MF 5.04.

NRC expects to close MF 5.04 (Certain Risk-Significant K_d Values for Saltstone) under PO §61.41 after NRC determines that model support for the sorption coefficients assumed for Ra and Se for saltstone is adequate. MF 5.04 may be closed based on DOE measurements on either field-emplaced or simulated saltstone. NRC could close MF 5.04 (and open a new MF for Se) if NRC determines that the inventory of Ra-226 and its ancestors is consistent with the revised inventory assumed in Case K under MFs 1.01 and 1.02.

3.5.5. MF 5.05: Potential for Short-Term Rinse-Release from Saltstone

Studies of radionuclide release from simulated saltstone demonstrated an initial relatively rapid radionuclide release that is characterized as a “rinse-release” phenomenon (see Tallent (1987) and Pabalan (2012)). That phenomenon was not in the DOE 2009 PA conceptual model.

That observed release may be due to the radionuclides not being fully encapsulated in the saltstone. For redox sensitive radionuclides (e.g., Tc), that release may also be due to it not being fully reduced in the saltstone. Alternatively, that release may be due to an experimental artifact related to the grinding of the simulated saltstone under atmospheric conditions containing oxygen. If that release represents radionuclides that are not encapsulated in or reduced by the saltstone, then a high release rate with resulting high dose could occur when the first few pore volumes of water through the saltstone reach the point of compliance. If water is excluded from the SDF for extended periods after site closure, then this rinse-release, if applicable to field-emplaced saltstone, would not occur until well after the sheet-drain system is grouted and closed.

NRC will monitor the development of support for the DOE assumption that short-term rinse release of radionuclides from saltstone seen in laboratory experiments will not significantly affect projected peak doses from groundwater pathways at the SDF.

NRC expects to close MF 5.05 (Potential for Short-Term Rinse-Release from Saltstone) under PO §61.41 after NRC determines that model support for the exclusion of rinse-release phenomenon from the conceptual model assumed in the DOE 2009 PA is adequate. Alternately, MF 5.05 may be closed after NRC determines that the phenomenon is well-understood and the effect on the projected dose is well supported.

3.6. MA 6 – Disposal Structure Performance

Disposal structure performance is important to the DOE conceptual model for SDF performance for two primary reasons. The first reason is that the large difference in hydraulic conductivity between the lower lateral drainage layer and the disposal structure roofs (for SDS 1 and SDS 4) or the HDPE/GCL composite layer (for disposal structures similar to SDS 2A) diverts most of the infiltrating water around the disposal structures. This diversion significantly reduces water flow through and radionuclide release from the disposal structures. The second reason is that the DOE modeled release of most radionuclides into the natural environment is significantly affected by retention in disposal structure concrete.

In DOE Cases K, K1, and K2 in the DOE 2009 PA, the peak dose from Tc is significantly delayed and reduced by the modeled Tc retention in and relatively gradual release from disposal structure concrete. In the NRC 2012 TER *(NRC, 2012)*, NRC staff determined that this degree of Tc retention in disposal structure concrete is inconsistent with the presence of fast pathways though fractures in the disposal structure concrete or joints between disposal structure components (e.g., floors, walls) and additional information was needed regarding the expected disposal structure performance.

The NRC staff expects that sorption of all radionuclides in disposal structure concrete will be limited as fast flow paths develop during the performance period. Fast pathways are expected to be important not only because of their potential effect on water flow, but also because radionuclides are expected to encounter significantly fewer sorption sites in fast pathways than they are in a cementitious matrix. NRC staff should evaluate the projected changes in sorption with time in response to disposal structure fracturing and degradation of non-cementitious materials.

The NRC monitored the performance of SDS 1 and SDS 4 as well as the construction of SDS 2A and SDS 2B during eight onsite observation visits (i.e., October 2007, March 2008, March 2009, June 2009, April 2010, July 2010, January 2011, April 2011), as documented in the reports and photographs. As such, NRC staff should be familiar with the following specific topics:

- SDS 4 fracturing and contaminated seeps *(NRC, 2008a; NRC, 2008b; NRC, 2011a)*;
- SDS 2A and SDS 2B floor construction *(NRC, 2009b)*;
- Deviations from the design of SDS 2A and SDS 2B wall joint closure strips with resulting increased potential for fast pathways *(NRC, 2009c)*;
- Disposal structure construction photographs from February 8-11, 2010, onsite observation visit;
- SDS 2A and SDS 2B damp spots prior to hydrostatic test and photographs *(NRC, 2010b)*; and
- Repairs and design changes made in response to the SDS 2A and SDS 2B hydrostatic test results *(NRC, 2010c; NRC, 2011b).*

As DOE prepares to construct and then constructs additional disposal structures, NRC will monitor any changes that DOE makes to disposal structure designs.

3.6.1. MF 6.01: Certain Risk-Significant K_d Values in Disposal Structure Concrete

Radionuclide sorption in disposal structure concrete is important to the DOE SDF conceptual model. In the NRC 2012 TER *(NRC, 2012)*, NRC staff determined that there was not an adequate basis for the K_d values DOE used to represent Se or Ra sorption in cementitious materials.

The dose from Ra-226 dominated the projected dose in Case A and in most of the DOE sensitivity analyses, other than Case K. DOE relied on measured values of Sr sorption as a surrogate for Ra sorption in the DOE 2009 PA. In Case K, DOE used a much lower revised inventory of Ra-226 and its ancestors, Th-230 and U-234. If the Case K inventory of those radionuclides is accurate, then it is unlikely Ra-226 will be significant to dose. However, if Ra-226 continues to make a significant contribution to the DOE projected dose, then it will be important to reduce the uncertainty in Ra sorption by collecting radionuclide-specific information about Ra sorption in disposal structure concrete. Regarding Se, it appears that the K_d values in the DOE 2009 PA for oxidizing disposal structure concrete were more applicable to selenite than the expected oxidized species, selenate.

The NRC will monitor the development of information about Ra and Se sorption in disposal structure concrete. MF 6.01 is currently limited to Se and Ra, but it may be expanded to include other radionuclides if their K_d values in disposal structure concrete prove to be risk-significant and are not adequately supported.

The K_d value for Tc in disposal structure concrete is covered in MF 6.02 because of its significant effect on the projected dose. Also, the sorption coefficients for radionuclides that have the potential to affect the dose, but are less risk-significant are covered in MF 10.06. MF 6.01 is also related to other MFs, where NRC staff should evaluate projected changes in sorption with time in response to disposal structure fracturing (MF 6.04) and degradation of non-cementitious materials (MF 6.05).

NRC expects to close MF 6.01 (Certain Risk-Significant K_d Values in Disposal Structure Concrete) under PO §61.41 after NRC determines that DOE information about Ra and Se sorption in disposal structure concrete is appropriate. Regarding Ra-226, that information could include either material-specific measurements of Ra sorption to disposal structure concrete or additional support for the revised lower inventory estimates for Ra-226 and Th-230 that DOE used in Case K. Regarding Se-79, that information could include additional model support (e.g., results of laboratory experiments) for the appropriate sorption coefficient for Se in oxidized disposal structure concrete. Alternately for Ra and Se, if the DOE dose projection changes, then NRC could determine that the potential dose from Ra and Se is appropriate without sorption in the disposal structure concrete. DOE may provide additional model support (e.g., results of laboratory experiments) to demonstrate that the sorption coefficient for Se in oxidized disposal structure concrete reflects the sorption of selenate rather than selenite. For either Ra-226 or Se-79, if appropriate information for one of those radionuclides is provided by DOE, but not the other radionuclide, then NRC could close MF 6.01 and open a new MF for the other radionuclide.

3.6.2. MF 6.02: Tc Sorption in Disposal Structure Concrete

NRC analyses of intermediate results from the DOE Case K PORFLOW model demonstrate the importance of Tc retention in disposal structure concrete to SDF performance. The DOE Cases K, K1, and K2, display unexpected reconcentration of Tc in the disposal structure floor and walls. For example, by approximately 13,000 years after site closure, over 90% of the Tc originally in the saltstone has been released and reconcentrated in the disposal structure concrete. This model behavior occurs because Tc sorption in the disposal structure floors and walls is modeled with much higher sorption coefficients than Tc sorption in oxidized saltstone. NRC staff does not believe this modeled behavior is realistic. SDF performance is sensitive to that sorption assumption. Thus, the NRC will monitor the development of information about Tc sorption in disposal structure concrete. NRC staff should evaluate whether the K_d values are consistent with the expected conditions to which disposal structure concrete may be exposed (e.g., presence of trace levels or more of oxygen). NRC staff should ensure that the values are consistent with (or conservatively bound the effects of) fractures and other potential fast pathways (see MFs 6.04 and 6.05). NRC staff should evaluate projected changes in sorption with time in response to disposal structure fracturing (MF 6.04) and degradation of non-cementitious materials (MF 6.05).

NRC expects to close MF 6.02 (Tc Sorption in Disposal Structure Concrete) under PO §61.41 after NRC determines that K_d values for Tc in reduced and oxidized disposal structure concrete are well-supported. Alternately, if the DOE dose projection changes, then NRC could determine that the potential dose from Tc is appropriate without Tc sorption in disposal structure concrete.

3.6.3. MF 6.03: Performance of Disposal Structure Roofs and HDPE/GCL Layers

In most of the DOE modeled cases, the lower lateral drainage layer above each disposal structure diverts nearly all of the infiltrating water around the disposal structures. This modeled diversion of infiltrating water was due to the large difference in the projected hydraulic conductivity of the highly conductive lower lateral drainage layer and the relatively impermeable disposal structure roofs (for SDS 1 and SDS 4) and the HDPE/GCL layer (for disposal structures similar to SDS 2A). An increase in the amount of infiltrating water will likely result in a higher dose to an off-site member of the public because an increase in the amount of infiltrating water will increase the amount of leaching from- and the rate of degradation of-the waste form. Thus, NRC will monitor model support for the long-term performance of the lower lateral drainage layer.

MF 6.03 involves NRC monitoring for all of the layers involved because the long-term performance of the lower lateral drainage layer depends on the contrast between high and low conductivity layers. NRC staff should evaluate information about processes that could reduce the conductivity of the highly permeable lower lateral drainage layer (e.g., clogging of the high-conductivity sand layer) and processes that could increase the assumed low hydraulic conductivity of the disposal structure roofs or HDPE/GCL layers overlying the disposal structures (for disposal structures similar to SDS 2A).

NRC expects to close MF 6.03 (Performance of Disposal Structure Roofs and HDPE/GCL Layers) under PO §61.41 after NRC determines that model support for the amount of water that DOE expects to be diverted by the lower lateral drainage layer, including support for the hydraulic conductivity of the relevant engineered layers, is sufficient. Alternately, NRC could close MF 6.03 if DOE conservatively assumes less diversion around the disposal structures in the PA model.

3.6.4. MF 6.04: Disposal Structure Concrete Fracturing

The DOE Cases K, K1, and K2 display unexpected reconcentration of Tc in the disposal structure floor and walls. That modeled behavior does not appear to account for the potential formation of fast pathways through the disposal structure floor and walls due to disposal structure concrete fracturing. Radionuclides that flow through fast pathways are not expected to experience as much sorption as radionuclides moving through an unfractured cementitious matrix because they are expected to encounter fewer sorption sites. Also, Tc sorption is expected to be further lessened in fractured saltstone (as compared to Tc sorption through an unfractured and chemically reducing saltstone) because fast pathways are likely to be more oxidizing than the matrix, due to increased contact with soil gas and dissolved oxygen in infiltrating water. The NRC will monitor the development of model support regarding the long-term fracturing of disposal structure concrete because of the importance of radionuclide sorption in the disposal structure floors and walls to SDF performance.

NRC expects to close MF 6.04 (Disposal Structure Concrete Fracturing) under PO §61.41 after NRC determines that support for the amount of fracturing of the disposal structure floor and walls expected to occur during the performance period is adequate or if NRC determines that the estimate that DOE uses in the PA model is conservative.

3.6.5. MF 6.05: Integrity of Non-cementitious Materials

Modeled radionuclide retention in disposal structure concrete significantly affects the DOE projected SDF performance. Potential fast pathways through the disposal structure concrete may undermine that physical and chemical barrier to radionuclide release. In addition to fractures in the disposal structure concrete, fast pathways also may form in joints between disposal structure components that DOE expects to be sealed with non-cementitious materials (e.g., epoxy, neoprene seals). Fast pathways through joints were observed during hydrotesting of SDS 2A and SDS 2B *(NRC, 2010b)*. Due to possible future design modifications, similar fast pathways may develop in disposal structures that may be hidden from view *(NRC, 2010c)*. The NRC will monitor the development of model support for the long-term physical integrity of non-cementitious materials used in disposal structure joints because of the importance of radionuclide retention in the disposal structure concrete to the DOE projected SDF performance.

NRC expects to close MF 6.05 (Integrity of Non-cementitious Materials) under PO §61.41 after NRC determines that support for the assumed performance of non-cementitious materials used in the disposal structures is adequate. For example, DOE may perform accelerated testing to estimate long-term performance. Alternately, DOE may be able to use a conservative estimate in the PA model.

3.7. MA 7 – Subsurface Transport

Subsurface transport modeling can significantly affect the projected dose. For example, modeled dilution and plume overlap affect projected groundwater radionuclide concentrations, which directly affect projected dose. Subsurface transport modeling can also affect the modeled travel time. This can affect the projected dose, especially for short- to moderate-half-life radionuclides which may be retained long enough to experience significant decay prior to reaching the point of compliance of 100 meters. The travel time of a radionuclide to the point of compliance can also affect whether the radionuclide reaches this point within the performance period.

The sorption coefficient (K_d value) that is assumed for subsurface soils can have a significant effect on the modeled transport time and dose. Therefore, the NRC will monitor the K_d values for subsurface soil that are significant to the projected dose. Additional MFs (i.e., MFs 10.09 – MF 10.13) relate specifically to the far-field hydrologic modeling and cannot be addressed without significant modeling effort, such as a revision to the PA. Those MFs were included under MA 10 (Performance Assessment Model Revisions) because NRC staff expects that they will not be closed until after the PA is updated.

3.7.1. MF 7.01: Certain Risk-Significant K_d Values in Site Sand and Clay

With the exception of Se, in the NRC 2012 TER *(NRC, 2012)*, NRC staff determined that the subsurface K_d values that were significant to the projected dose were well supported. For Se in sand and clay, NRC staff determined that the K_d value of 1000 mL/g that DOE assumed in the DOE 2009 PA was not adequately supported because that value was representative of Se sorption in a low pH soil. The NRC will monitor the K_d values for Se in SRS sand and clay. MF 7.01 is currently limited to Se; however, if changes to other aspects of the disposal system lead to subsurface K_d values for additional elements becoming risk-significant, then MF 7.01 will be expanded to include those risk-significant elements.

Se was not identified as causing a significant portion of the dose in the DOE Case A or alternate deterministic sensitivity cases. DOE provided the results of a sensitivity analysis performed using the probabilistic GoldSim® model in which the Se K_d values for the sandy and clayey soils was 0 mL/g (SRR-CWDA-2010-00033). The results of that sensitivity analysis were that changing the Se K_d values resulted in only a small absolute increase in dose, but a large relative increase in the dose derived from Se-79. Combined with the uncertainty in other key parameters related to Se release and transport (i.e., MFs 5.04 and 6.01), NRC staff determined that the soil K_d values for Se may be important to dose.

NRC expects to close MF 7.01 (Certain Risk-Significant K_d Values in Site Sand and Clay) under PO §61.41 after NRC determines that site-specific measurements for the K_d value for Se in sand and clay are appropriate. Those measurements should consider the potential effect of the higher pH conditions that are likely to exist downgradient of the SDF. Alternatively, MF 7.01 may be closed if NRC determines that Se K_d values for SRS sand and clay do not have the potential to significantly affect the dose to an off-site member of the public. That determination should consider the uncertainty in other key parameters related to Se release and transport (i.e., MFs 5.04 and 6.01).

3.8. MA 8 – Environmental Monitoring

DOE conducts an effluent monitoring and environmental surveillance program on an ongoing basis at SRS. The data obtained through that program are summarized in an annual environmental report. A variety of environmental media, including groundwater; surface water; rainwater; air; vegetation; deer and hog meat; and soil, are monitored through that program. In assessing compliance with §61.41, the most useful environmental data to monitor is the groundwater data from the Z-Area because NRC staff expects that the groundwater will be the dominant pathway for long-term releases from the SDF. Increased concentrations of radionuclides or other saltstone indicators (e.g., nitrate) in the groundwater samples obtained near the SDF may indicate that radionuclides are leaching from the disposal structures and that the SDF is not performing as expected. Leak detectors installed beneath the disposal structures will also provide important information regarding the early performance of the saltstone waste form and disposal structures. The usefulness of other environmental data is somewhat limited because most of those samples are not obtained directly in the vicinity of the SDF and there are other potential sources of radioactivity at SRS, which makes it difficult to determine whether any observed concentration increases are attributable to the waste disposed of at the SDF. The NRC will focus on monitoring the leak detection systems and groundwater monitoring.

3.8.1. MF 8.01: Leak Detection

The DOE Consent Order of Dismissal with the SCDHEC requires DOE to install a system beneath SDS 3A, and every fifth cell constructed thereafter, to sample liquids, if any, for radionuclides and hazardous constituents. That consent order applied to the disposal of Deliquification, Dissolution, and Adjustment (DDA) waste, which is now complete. Therefore, that consent order no longer applies. However, NRC expects that this type of leak detection system at the SDF would provide extremely useful information regarding the early performance of the saltstone waste form, including whether or not any early failure of disposal structures has occurred. NRC staff will review the sample data from that system and will review whether the leak detection system was adequate for detecting leaks from SDS 3A.

NRC expects to close MF 8.01 (Leak Detection) under PO §61.41 after the leak detection system ends operation or after final waste disposal occurs, whichever comes later.

3.8.2. MF 8.02: Groundwater Monitoring

NRC staff expects that groundwater monitoring data may provide information regarding early release of radionuclides from saltstone. That data may also provide other indicators of performance, such as unexpected plumes of nitrate or increased alkalinity.

NRC will monitor the DOE groundwater monitoring program. A key aspect of that DOE groundwater monitoring program is the placement of the wells. For the wells to provide useful information, they must be located downgradient of the disposal structures and must be close enough to the disposal structures to see radionuclides or other indicators that may leach from the saltstone waste form. Similarly, it is important for the wells used to obtain information regarding the natural groundwater composition (i.e., “background” wells) to be upgradient of the disposal structures. The groundwater divide on the SDF may complicate the assessment of the well locations, especially because there is some uncertainty in the location of the divide and the location can change with changes to infiltration.

Other aspects of the groundwater monitoring program that NRC staff should consider include the sampling method and frequency and the analytical methods used for the samples. During onsite observation visits in October 2007 and March 2008, as documented in the reports *(NRC 2008a, NRC 2008b)*, NRC staff reviewed aspects of the DOE groundwater monitoring program. NRC staff will further evaluate the groundwater monitoring program in detail as new disposal structures and wells are constructed or if there is evidence that the performance is worse than expected (e.g., if environmental data shows that saltstone leaching is occurring or extensive waste form cracking is observed).

NRC will also monitor the groundwater monitoring data. The groundwater concentrations of radionuclides and saltstone indicators should be tracked by NRC staff over time to determine whether there are any trends in the data. Any observations of increased radioactivity, nitrate, pH, alkalinity, or other saltstone indicators should be followed up with additional sampling and analysis, and the source of these increased levels should be determined. NRC staff should consider historical concentrations measured in the groundwater and measured concentrations in upgradient wells when establishing background levels. 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 does not expect to close MF 8.02 (Groundwater Monitoring) under PO §61.41 because NRC will monitor groundwater data for the duration of NRC monitoring at the SDF.

3.9. MA 9 – Site Stability

In addition to being directly applicable to maintaining compliance with §61.44, site stability is important in limiting the infiltration through the SDF, which is important to SDF performance because of the importance of the hydraulic isolation of saltstone.

3.9.1. MF 9.01: Settlement Due to Increased Overburden

DOE studies documented in the reports from 2006 (K-ESR-Z-00001) and 2009 (K-ESR-Z-00002), projected more settlement due to overburden than addressed in the DOE 2009 PA. The NRC will monitor settlement due to increased overburden because of the potential for increased settlement to increase infiltration into the site.

NRC expects to close MF 9.01 (Settlement Due to Increased Overburden) under PO §61.41 after NRC determines that the projections of settlement in the recent geotechnical investigations will not adversely affect SDF performance. Alternately, DOE may provide NRC information that allows NRC to determine that the new DOE settlement projections are consistent with the values assumed in the DOE 2009 PA.

3.9.2. MF 9.02: Settlement Due to Dissolution of Calcareous Sediment

Site instability due to dissolution of calcareous sediment could affect the infiltration of water through the closure cap, disposal structures, and saltstone waste. Because of the importance of the hydraulic isolation of saltstone, NRC will monitor the DOE development of additional information to support the DOE conclusion that the potential for sink formation and the potential effects of sink formation are limited. NRC staff should evaluate the DOE assumptions regarding the potential ongoing dissolution of calcareous sediment. NRC staff should evaluate any new information that DOE provides to support the determination of whether or not reasonably projected future dissolution of calcareous sediment is significant to site stability.

NRC expects to close MF 9.02 (Settlement Due to Dissolution of Calcareous Sediment) under PO §61.41 after NRC assesses a new DOE projection of the likelihood of the formation of sinks during the period of performance at the SDF and any resulting effects on site stability.

3.10. MA 10 – Performance Assessment Model Revisions

DOE Manual 435.1-1, Change 1, requires that a PA and composite analysis be maintained to evaluate changes that could affect the performance, design, and operating bases for facilities. DOE prepares an annual PA maintenance program implementation plan that summarizes activities related to all the NDAA-related SRS facilities (i.e., FTF, HTF, SDF). The FY13 plan is in the DOE document SRR-CWDA-2013-00049, Rev. 1.

In the review of the DOE 2009 PA, NRC staff identified concerns with the PA that NRC staff expects will not be resolved until DOE revises the PA model under the DOE PA maintenance program. Those items are identified as MFs 10.01 through 10.13.

3.10.1. MF 10.01: Implementation of Conceptual Models

In the NRC 2012 TER *(NRC, 2012)*, NRC staff described the importance of implementation of the PA conceptual model. It is important for modeling codes to appropriately represent the conceptual models. NRC staff identified inconsistencies between the DOE intermediate models results and the conceptual models. NRC staff identified some potential errors in the DOE probabilistic model, which limited the usefulness of that model. NRC staff identified that some of the uncertainty distributions assumed by DOE in the probabilistic model were not adequately supported.

NRC will monitor any changes to the DOE 2009 PA, including the implementation of the conceptual models, consistency of any intermediate model results with the conceptual models, quality assurance of models and codes used, appropriate selection of parameter values, and the appropriate use of probabilistic factors, if used.

NRC expects to close MF 10.01 (Implementation of Conceptual Models) under PO §61.41 after DOE updates the PA and NRC determines that intermediate model results are consistent with the conceptual models, quality assurance methods used are appropriate, and parameter values and uncertainty ranges are appropriate.

3.10.2. MF 10.02: Defensibility of Conceptual Models

Uncertainty in conceptual models is difficult to capture in dose models, but it can dominate the uncertainty in dose projections. For example, an alternate conceptual model in which saltstone oxidizes for a long period of time during which little or no water flows into the waste and then is suddenly exposed to increased water flow (e.g., through HDPE failure) could generate a much larger peak dose than a more gradual failure. NRC will monitor the DOE consideration of alternate conceptual models in future PA development because of the potential importance of alternate conceptual models to dose projections. NRC will monitor the defensibility of the DOE conceptual models for releases of radionuclides and potential exposures to off-site members of the public.

NRC expects to close MF 10.02 (Defensibility of Conceptual Models) under PO §61.41 after DOE updates the PA and NRC determines that the conceptual models are appropriate.

3.10.3. MF 10.3: Diffusivity in Degraded Saltstone

Both DOE Case K results in the DOE 2009 PA and NRC analyses in the NRC 2012 TER *(NRC, 2012)* demonstrate the sensitivity of the magnitude and timing of the dose from Tc-99 to the rate of saltstone oxidation. In DOE Case K, the movement of an oxidation front is modeled as a function of the square root of time, which limits the progression of oxidation from older fractures. However, other functional relationships are possible if saltstone degrades and the diffusivity increases with time. NRC will monitor measurements of diffusivity in degraded saltstone and model support for assumptions about diffusivity in saltstone used in DOE updates to the PA.

NRC expects to close MF 10.03 (Diffusivity in Degraded Saltstone) under PO §61.41 after DOE updates the PA and NRC determines that the diffusivity information, including the model of the movement of the oxidation front, is well-supported.

3.10.4. MF 10.04: K_d Values for Saltstone

Based on DOE and NRC sensitivity analyses for Case A and Case K, NRC staff expects that K_d values for saltstone will significantly affect projected doses. NRC will monitor changes in K_d values in saltstone for radionuclides that could potentially become risk-significant that were not risk-significant in the DOE 2009 PA and NRC 2012 TER *(NRC, 2012)*. NRC will monitor model support for K_d values in saltstone for radionuclides that become risk-significant in the updated PA. NRC staff should consider that K_d values in saltstone and the uncertainty in those values can affect which radionuclides are the primary dose contributors. The radionuclides that are risk-significant in the DOE 2009 PA are already under other MFs (i.e., Tc under MF 5.02, Ra and Se under MF 5.04).

NRC expects to close MF 10.04 (K_d Values for Saltstone) under PO §61.41 after DOE updates the PA and NRC determines that the K_d values for saltstone for any radionuclides that become risk-significant in the updated PA are well-supported.

3.10.5. MF 10.05: Moisture Characteristic Curves

The DOE Case A and sensitivity analyses in the DOE 2009 PA rely on moisture characteristic curves (MCCs). Based on comparison to MCCs published for similar materials, the MCCs in DOE Case A reduce water flow through saltstone, fracture, and disposal structure concrete more than expected. The DOE Case K analysis does not take credit for decreasing permeability with decreasing saturation (i.e. as described by MCCs). Rather, the DOE Case K analysis assumes the relative permeability is always 1 (i.e., independent of saltstone saturation). NRC staff expects the DOE development of appropriate model support for any MCCs to be important because of the sensitivity of projected doses to the modeled permeability of saltstone. NRC will monitor the DOE development of model support for any MCCs used to modify the permeability of saltstone or disposal structure concrete in any updated PA.

NRC expects to close MF 10.05 (Moisture Characteristic Curves) under PO §61.41 after DOE updates the PA and NRC determines that the MCCs are well-supported. Alternatively, MF 10.05 may be closed if, in an updated PA, DOE assumes the relative permeability is 1, which means that DOE does not use MCCs in the updated PA.

3.10.6. MF 10.06: K_d Values for Disposal Structure Concrete

NRC staff expects that K_d values for disposal structure concrete can have a significant effect on the modeled release rate of radionuclides into the near-field environment. Release rates can directly affect the projected dose. NRC will monitor model support for K_d values used to represent sorption of radionuclides in disposal structure concrete. NRC will monitor changes in K_d values for disposal structure concrete for radionuclides that could potentially become risk-significant that were not risk-significant in the DOE 2009 PA and NRC 2012 TER *(NRC, 2012)*. NRC will monitor model support for K_d values for disposal structure concrete for radionuclides that become risk-significant in the updated PA. NRC staff should consider that K_d values in structure concrete and the uncertainty in those values can affect which radionuclides are the primary dose contributors. The radionuclides that are risk significant in the DOE 2009 PA are already under other MFs (i.e., Tc under MF 6.02, Se and Ra under MF 6.01).

NRC expects to close MF 10.06 (K_d Values for Disposal Structure Concrete) under PO §61.41 after DOE updates the PA and NRC determines that the K_d values for disposal structure concrete for any radionuclides that become risk-significant in the updated PA are well-supported.

3.10.7. MF 10.07: Calculation of Build-Up in Biosphere Soil

In the NRC 2012 TER *(NRC, 2012)*, NRC staff determined that the K_d values assumed by DOE for surface soils in the soil build-up calculations may have resulted in an underestimation of concentration of radionuclides in the biosphere soil, which would lead to an underestimation of the projected dose from ingestion of plant and animal products. For example, the K_d values that DOE used in the build-up analysis were based on conservative hydrologic transport modeling values (i.e., values were purposefully biased low). However, that selection process is non-conservative when applied to irrigation and soil sorption modeling because lower sorption values could underestimate radionuclide build-up. NRC staff scoping calculations demonstrated that using site-specific K_d measurements could increase estimated build-up by approximately two to five times. NRC will monitor whether or not K_d values for key radionuclides in surface soil significantly increase projected radionuclide build-up in biosphere soil. NRC staff should evaluate the potential effect on projected dose.

NRC expects to close MF 10.07 (Calculation of Build-Up in Biosphere Soil) under PO §61.41 after DOE updates the PA and NRC determines that the soil K_d values are well-supported in the soil build-up calculation (i.e., if DOE chose conservative low K_d values in the transport calculation, then the soil K_d values may not be the same K_d values used in the transport calculation).

3.10.8. MF 10.08: Consumption Factors and Uncertainty Distributions for Transfer Factors

In the NRC 2012 TER *(NRC, 2012)*, NRC concluded that some of the parameter values that DOE selected for consumption factors and uncertainty distributions for transfer factors may have resulted in an underestimation of dose. For example, in the DOE 2009 PA, DOE assumed consumption of drinking water to be approximately 1 liter/day, half of the EPA recommended assumption of 2 liters/day, which was not supported by site-specific information. The calculated dose is strongly dependent on the drinking water intake rate because a large portion of the estimated dose from the SDF is from the drinking water pathway. Also, DOE did not consider uncertainty distributions for the transfer factors in the biosphere calculations. Transfer factors can have considerable uncertainty, which needs to be evaluated in future probabilistic models. NRC will monitor consumption factors and uncertainty distributions for transfer factors.

NRC expects to close MF 10.08 (Consumption Factors and Uncertainty Distributions for Transfer Factors) under PO §61.41 after DOE updates the PA and NRC determines that the values of consumption factors and uncertainty distributions for transfer factors are well-supported.

3.10.9. MF 10.09: K_d Values for SRS Soil

The sorption of radionuclides onto subsurface soil can affect the dose to an off-site member of the public. NRC will monitor changes in soil K_d values for radionuclides that could potentially become risk-significant that were not risk-significant in the DOE 2009 PA and NRC 2012 TER *(NRC, 2012)*. NRC will monitor model support for site-specific K_d values for radionuclides that become risk-significant in the updated PA. NRC staff should consider that K_d values for SRS soil and the uncertainty in those values can affect which radionuclides are the primary dose contributors. The radionuclides that are risk-significant in the DOE 2009 PA are already covered under other MFs.

NRC expects to close MF 10.09 (K_d Values for SRS Soil) under PO §61.41 after DOE updates the PA and NRC determines that the site-specific K_d values for any radionuclides that become risk-significant in the updated PA are well-supported.

3.10.10. MF 10.10: Far-Field Model Calibration

In the NRC 2012 TER *(NRC, 2012),* NRC staff described that improved DOE model calibration, particularly in the area near the SDF, would provide additional support for the DOE assumed level of dilution and dispersion of SDF sources. Based on potentially greater plume overlap and less aquifer dilution than assumed in the DOE 2009 PA, slight changes to the location of the groundwater divide or hydraulic gradients could result in increased dose projections. NRC will monitor the adequacy of the DOE far-field model calibration.

NRC expects to close MF 10.10 (Far-Field Model Calibration) under PO §61.41 after DOE updates the PA and NRC determines that the far-field model calibration, particularly in the area near the SDF, is adequate.

3.10.11. MF 10.11: Far-Field Model Source Loading Approach

In the NRC 2012 TER *(NRC, 2012)*, NRC scoping simulations using Tc-99 fluxes from Case K indicated that, based on the approach for far-field source loading, the peak sector concentrations at the 100 meter boundary could be significantly higher for some sectors. Specifically, projected doses could increase if DOE ensures that the same amount of mass is loaded in the saturated zone underneath each disposal structure and DOE ensures that the source loading occurs at the water table. NRC will monitor the DOE source loading approach so that DOE does not significantly underproject the dose estimates.

NRC expects to close MF 10.11 (Far-Field Model Source Loading Approach) under PO §61.41 after DOE updates the PA and NRC determines that the far-field source loading approach in the model is adequate.

3.10.12. MF 10.12: Far-Field Model Dispersion

The DOE results in the FTF RAI responses (SRR-CWDA-2009-00054) indicated that additional grid refinement may be necessary to reduce numerical dispersion in cases of very low to no assumed physical dispersion. NRC staff expects that analysis to be relevant to the SDF because the local FTF and SDF models have the same grid resolution. For example, if no physical dispersion is assumed, then the peak concentrations associated with a pulse release of a conservative tracer would be a factor of approximately three to four times higher with a grid refined by a factor of two in each dimension (i.e., eight times more elements). NRC will monitor the appropriateness of selected dispersivities and the need for additional vertical or horizontal mesh refinement to evaluate whether or not the contaminant plumes are artificially dispersed in the far-field model.

NRC expects to close MF 10.12 (Far-Field Model Dispersion) under PO §61.41 after DOE updates the PA and NRC determines that the grid refinement used in any hydrological model supporting the updated PA does not increase modeled dispersion beyond the expected physical dispersion.

3.10.13. MF 10.13: Impact of Calcareous Zones on Contaminant Flow and Transport

In the DOE 2009 PA, many of the contaminant pathways were modeled as traversing the lower zone of the Upper Three Runs (UTR) aquifer, where calcareous materials are more pervasive in the subsurface at SRS. Some evidence indicates that contaminants in the Burial Ground Complex (i.e., located in E-Area at the General Separations Area (GSA)) and the Chemical, Metals, and Pesticide Pits (i.e., located off the GSA) may be preferentially transported within those zones (see Thayer (1995)). If DOE identifies calcareous zone seeps, then tracer studies in the UTR-lower zone using innocuous tracers could be conducted to better understand the effect of those zones on contaminant flow and transport. NRC will monitor the DOE effort to collect data and information to evaluate the potential impact of calcareous zones on contaminant flow and transport.

NRC expects to close MF 10.13 (Impact of Calcareous Zones on Contaminant Flow and Transport) under PO §61.41 after DOE investigates potential preferential pathways due to subsurface calcareous zones and NRC determines that the DOE representation of any preferential pathways due to calcareous zones is adequate.

(Intentionally Left Blank)

4. MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.42

§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 NRC will continue to assess DOE compliance with the §61.42 PO by evaluating the DOE intruder analysis. A specific dose limit is not in the PO. However, a dose limit of 5 mSv/yr (500 mrem/yr) was used in the draft environmental impact statement for 10 CFR Part 61 for development of the waste classification requirements *(NRC, 1981)*, which were developed to provide for inadvertent intruder protection. As described in NUREG-1854 *(NRC, 2007b)*, NRC staff uses that 5 mSv/yr (500 mrem/yr) dose limit in evaluating intruder scenarios.

In the NRC 2012 TER *(NRC, 2012)*, NRC concluded with reasonable assurance that the dose to an inadvertent intruder is likely to be below 5 mSv/yr (500 mrem/yr), based on the DOE Case K, K1, and K2 analyses, although the projected intruder dose was a significant fraction of the 5 mSv/yr (500 mrem/yr) limit. Therefore, key assumptions in the DOE analysis must be correct for NRC to continue to have reasonable assurance that §61.42 will be met. Alternatively, DOE can incorporate new information into a revised intruder analysis.

The pathways assumed in the chronic intruder agriculture calculation for §61.42 were the same as those assumed for the member of the public for §61.41, except that the groundwater concentrations were based on different well locations (i.e., on the SDF instead of outside the 100 meter boundary). In the intruder assessment for §61.42, DOE assumed that the well will not go directly through the disposal structures because the disposal structures contain long-lasting materials that are clearly distinguishable from the surrounding soil. DOE determined that because the local soil is generally sandy, local well drillers do not expect to need to drill through high-strength geologic materials when constructing a drinking water well. Consequently, the well driller would stop operations and move to a different location upon encountering engineered barriers, such as the closure cap erosion barrier or the disposal structure concrete roof. Although DOE does not consider the construction of a well through saltstone to be a credible scenario, DOE performed two sensitivity analyses to assess the potential dose resulting from drilling directly into a disposal structure and determined that the dose from the groundwater pathway bounds the dose from drill cuttings. In the NRC 2012 TER, NRC determined that the scenarios and pathways that DOE considered for an inadvertent intruder were appropriate, based on the regional practices near SRS.

The dose for both the off-site member of the public for §61.41 and the intruder for §61.42 were based primarily on the groundwater pathway. Therefore, most of the MFs identified as being important to compliance with §61.41 for the member of the general public are also applicable to §61.42 for the inadvertent intruder. One difference is that the radionuclides that are relatively short-lived and are slowly transported (e.g., Sr-90) may be significant to the intruder for §61.42, but are not significant to a member of the public because those radionuclides are projected to decay before they reach the 100 meter boundary. Also, items related to the saturated zone transport are typically not applicable to the intruder assessment for §61.42 because the receptor is assumed to be consuming water from directly below or adjacent to the disposal structure. If key assumptions in the groundwater modeling change then this could result in items related to the far-field transport becoming important to the projected dose to an inadvertent intruder.

As indicated in the Executive Summary, information in one chapter of this monitoring plan that is the same as an earlier chapter will not be repeated in the latter chapter. Much of the information in the MAs and MFs in Chapter 3 (Monitoring to Assess Compliance with 10 CFR 61.41) is the same as the information in Chapter 4 (Monitoring to Assess Compliance with 10 CFR 61.42) and so, that same information also applies to Chapter 4, but will not be repeated in Chapter 4.

4.1. MA 1 – Inventory

The information in Section 4.1 (MA 1 – Inventory) for §61.42 is similar to the information in Section 3.1 (MA 1 – Inventory) for §61.41. In addition to the radionuclides identified as potentially risk-significant to a member of the public (i.e., Tc-99, I-129, Ra-226, Se-79, ancestors of Ra-226), Sr-90 may also be risk-significant to the inadvertent intruder.

4.1.1. MF 1.01: Inventory in Disposal Structures

The distribution of inventory among the disposal structures will be important in determining potential doses to an inadvertent intruder because a single disposal structure can dominate the dose to an inadvertent intruder. Each disposal structure inventory needs to be consistent with assumptions in the most recent DOE PA so that NRC can continue to conclude with reasonable assurance that DOE disposal actions meet §61.42.

NRC will monitor the inventory in each disposal structure in comparison to the values in Table A-1 (Projected Inventory at Time of Closure) of this monitoring plan. As discussed in Section 3.1.1 for §61.41, if the inventory of a radionuclide in a particular disposal structure is higher than the values in that table, then an NRC analysis should be performed to understand the dose consequences of the increased inventory. Although the dose to an inadvertent intruder is expected to be dominated by an individual disposal structure, the plume from neighboring disposal structures may contribute to the dose and should be considered as part of that analysis. For more information about inventory in disposal structures, see Section 3.1.1 (MF 1.01 – Inventory in Disposal Structures).

NRC expects to close MF 1.01 (Inventory in Disposal Structures) under PO §61.42 after DOE has completed waste disposal at the SDF and determined the final inventory in each disposal structure.

4.1.2. MF 1.02: Methods Used to Assess Inventory

NRC will monitor the methods DOE uses to assess radionuclide inventories because of the considerable uncertainty in inventory estimates and the importance of radionuclide inventory to dose to an inadvertent intruder. In addition to the radionuclides specified in Section 3.1.2 for §61.41, under §61.42 NRC will also monitor the inventory of Sr-90. For more information about methods used to assess inventory, see Section 3.1.2 (MF 1.02 – Methods Used to Assess *Inventory).*

NRC expects to close MF 1.02 (Methods Used to Assess Inventory) under PO §61.42 after DOE has completed waste disposal at the SDF and determined the final inventory in each disposal structure.

4.2. MA 2 – Infiltration and Erosion Control

The closure cap is important to intruder protection primarily because the erosion barrier is expected to discourage inadvertent intrusion by presenting a physical barrier and maintaining cover depth. In addition, the cover is expected to limit radionuclide releases to groundwater by limiting groundwater infiltration to the site, especially in the first few centuries after site closure, when the potential risk to an inadvertent intruder from short-lived radionuclides is the greatest. For more information about how the closure cap relates to infiltration and erosion control, see Section 3.2 (MA 1 – Infiltration and Erosion Control).

4.2.1. MF 2.01: Hydraulic Performance of Closure Cap

The information in Section 4.2.1 (MF 2.01 – Hydraulic Performance of Closure Cap) for §61.42 is the same as the information in Section 3.2.1 (MF 2.01 – Hydraulic Performance of Closure Cap) for §61.41.

NRC expects to close MF 2.01 (Hydraulic Performance of Closure Cap) under PO §61.42 after NRC determines that the hydraulic performance of the as-built closure cap is adequate. Given the importance of construction activities on the performance of the cap, MF 2.01 will not be closed prior to construction of the cap.

4.2.2. MF 2.02: Erosion Protection

DOE needs more model support to demonstrate that the physical stability of the final closure cap is consistent with the assumed performance of the cap in the DOE 2009 PA. NRC staff will evaluate preliminary erosion protection designs, any significant changes to the design before construction, construction quality, and information related to the physical stability of the vegetative and topsoil layers. For more information about how the closure cap relates to erosion control, see Section 3.2.2 (MF 2.02 – Erosion Protection).

NRC expects to close MF 2.02 (Erosion Protection) under PO §61.42 after NRC determines that the physical stability of the final closure cap is adequate. Given the importance of construction activities on the performance of the cap, MF 2.02 will not be closed prior to construction of the cap.

4.3. MA 3 – Waste Form Hydraulic Performance

The information in Section 4.3 (MA 3 – Waste Form Hydraulic Performance) for §61.42 is the same as the information in Section 3.3 (MA 3 – Waste Form Hydraulic Performance) for §61.41.

4.3.1. MF 3.01: Hydraulic Conductivity of Field-Emplaced Saltstone

The information in Section 4.3.1 (MF 3.01 – Hydraulic Conductivity of Field-Emplaced Saltstone) for §61.42 is the same as the information in Section 3.3.1 (MF 3.01 – Hydraulic Conductivity of Field-Emplaced Saltstone) for §61.41.

NRC expects to close MF 3.01 (Hydraulic Conductivity of Field-Emplaced Saltstone) under PO §61.42 after NRC determines that model support for the saturated hydraulic conductivity of field-emplaced saltstone is sufficient.

4.3.2. MF 3.02: Variability of Field-Emplaced Saltstone

The information in Section 4.3.2 (MF 3.02 – Variability of Field-Emplaced Saltstone) for §61.42 is the same as the information in Section 3.3.2 (MF 3.02 – Variability of Field-Emplaced Saltstone) for §61.41.

NRC expects to close MF 3.02 (Variability of Field-Emplaced Saltstone) under PO §61.42 after NRC determines that saltstone production, placement, and curing conditions that significantly affect saltstone hydraulic properties are well controlled.

4.3.3. MF 3.03: Applicability of Laboratory Data to Field-Emplaced Saltstone

The information in Section 4.3.3 (MF 3.03 – Applicability of Laboratory Data to Field-Emplaced Saltstone) for §61.42 is the same as the information in Section 3.3.3 (MF 3.03 – Applicability of Laboratory Data to Field-Emplaced Saltstone) for §61.41.

NRC expects to close MF 3.03 (Applicability of Laboratory Data to Field-Emplaced Saltstone) under PO §61.42 after NRC determines that representing the hydraulic properties of field-emplaced saltstone with the hydraulic properties of laboratory-produced samples is adequate. That assessment should account for the range of expected disposal conditions of field-emplaced saltstone as well as effects of scale. Alternately, MF 3.03 may be closed if NRC determines that DOE bases the hydraulic properties of saltstone on the properties of an appropriate range of samples of field-emplaced saltstone, rather than on measurements of laboratory-produced samples.

4.3.4. MF 3.04: Effect of Curing Temperature on Saltstone Hydraulic Properties

The information in Section 4.3.4 (MF 3.04 – Effect of Curing Temperature on Saltstone Hydraulic Properties) for §61.42 is the same as the information in Section 3.3.4 (MF 3.04 – Effect of Curing Temperature on Saltstone Hydraulic Properties) for §61.41.

NRC expects to close MF 3.04 (Effect of Curing Temperatures on Saltstone Hydraulic Properties) under PO §61.42 after NRC determines that projected SDF performance is based on estimates of the hydraulic properties of saltstone (e.g., hydraulic conductivity and diffusivity) that are well-supported. That support should account for the range of curing conditions (i.e., temperatures values, humidity values) experienced by field-emplaced saltstone.

4.4. MA 4 – Waste Form Physical Degradation

The information in Section 4.4 (MA 4 – Waste Form Physical Degradation) for §61.42 is the same as the information in Section 3.4 (MA 4 – Waste Form Physical Degradation) for §61.41.

4.4.1. MF 4.01: Waste Form Matrix Degradation

The information in Section 4.4.1 (MF 4.01 – Waste Form Matrix Degradation) for §61.42 is the same as the information in Section 3.4.1 (MF 4.01 – Waste Form Matrix Degradation) for §61.41.

NRC expects to close MF 4.01 (Waste Form Matrix Degradation) under PO §61.42 after NRC determines that support for modeled changes in the saturated hydraulic conductivity and diffusivity during the performance period is sufficient.

4.4.2. MF 4.02: Waste Form Macroscopic Fracturing

The information in Section 4.4.2 (MF 4.02 – Waste Form Macroscopic Fracturing) for §61.42 is the same as the information in Section 3.4.2 (MF 4.02 – Waste Form Macroscopic Fracturing) for §61.41.

NRC expects to close MF 4.02 (Waste Form Macroscopic Fracturing) under PO §61.42 after NRC determines that model support for the assumed formation of macroscopic fractures during the performance period is sufficient.

4.5. MA 5 – Waste Form Chemical Performance

The information in Section 4.5 (MA 5 – Waste Form Chemical Performance) for §61.42 is the same as the information in Section 3.5 (MA 5 – Waste Form Chemical Performance) for §61.41.

4.5.1. MF 5.01: Radionuclide Release from Field-Emplaced Saltstone

The information in Section 4.5.1 (MF 5.01 – Radionuclide Release from Field-Emplaced Saltstone) for §61.42 is the same as the information in Section 3.5.1 (MF 5.01 – Radionuclide Release from Field-Emplaced Saltstone) for §61.41.

NRC expects to close MF 5.01 (Radionuclide Release from Field-Emplaced Saltstone) under PO §61.42 after NRC determines that measurements of radionuclide release rates from field-emplaced saltstone used in the PA are reliable.

4.5.2. MF 5.02: Chemical Reduction of Tc by Saltstone

The information in Section 4.5.2 (MF 5.02 – Chemical Reduction of Tc by Saltstone) for §61.42 is the same as the information in Section 3.5.2 (MF 5.02 – Chemical Reduction of Tc by Saltstone) for §61.41.

NRC expects to close MF 5.02 (Chemical Reduction of Tc by Saltstone) under PO §61.42 after NRC determines that: (1) model support for the chemical reduction of Tc(VII) to Tc(IV) is robust; and (2) this reduced state is maintained under field conditions. NRC expects that DOE will inform NRC what the ranges of those conditions are expected to be during the performance period.

4.5.3. MF 5.03: Reducing Capacity of Saltstone

The information in Section 4.5.3 (MF 5.03 – Reducing Capacity of Saltstone) for §61.42 is the same as the information in Section 3.5.3 (MF 5.03 – Reducing Capacity of Saltstone) for §61.41.

NRC expects to close MF 5.03 (Reducing Capacity of Saltstone) under PO §61.42 after NRC determines that information for the initial reducing capacity of saltstone and the expected evolution of redox conditions over time is adequate.

4.5.4. MF 5.04: Certain Risk-Significant K_d Values for Saltstone

The potential dose is dependent on the ability of the saltstone to retain risk-significant radionuclides. In addition to the radionuclides specified in Section 3.5.4 for §61.41, under

§61.42 NRC will also monitor Sr. For more information about certain risk-significant K_d values for saltstone, see Section 3.5.4 (MF 3.04 – Certain Risk-Significant K_d Values for Saltstone).

NRC expects to close MF 5.04 (Certain Risk-Significant K_d Values for Saltstone) under PO §61.42 after NRC determines that model support for the sorption coefficients assumed for Ra and Se for saltstone is adequate. MF 5.04 may be closed based on DOE measurements on either field-emplaced or simulated saltstone. NRC could close MF 5.04 (and open a new MF for Se) if NRC determines that the inventory of Ra-226 and its ancestors is consistent with the revised inventory assumed in Case K under MFs 1.01 and 1.02.

4.5.5. MF 5.05: Potential for Short-Term Rinse-Release from Saltstone

The information in Section 4.5.5 (MF 5.05 – Potential for Short-Term Rinse-Release from Saltstone) for §61.42 is the same as the information in Section 3.5.5 (MF 5.05 – Potential for Short-Term Rinse-Release from Saltstone) for §61.41.

NRC expects to close MF 5.05 (Potential for Short-Term Rinse-Release from Saltstone) under PO §61.42 after NRC determines that model support for the exclusion of rinse-release phenomenon from the conceptual model assumed in the DOE 2009 PA is adequate. Alternately, MF 5.05 may be closed after NRC determines that the phenomenon is well-understood and the effect on the projected dose is well supported.

4.6. MA 6 – Disposal Structure Performance

The information in Section 4.6 (MA 6 – Disposal Structure Performance) for §61.42 is the same as the information in Section 3.6 (MA 6 – Disposal Structure Performance) for §61.41.

4.6.1. MF 6.01: Certain Risk-Significant K_d Values in Disposal Structure Concrete

The information in Section 4.6.1 (MF 6.01 – Certain Risk-Significant K_d Values in Disposal Structure Concrete) for §61.42 is the same as the information in Section 3.6.1 (MF 6.01 – Certain Risk-Significant K_d Values in Disposal Structure Concrete) for §61.41.

NRC expects to close MF 6.01 (Certain Risk-Significant K_d Values in Disposal Structure Concrete) under PO §61.42 after NRC determines that DOE information about Ra and Se sorption in disposal structure concrete is appropriate. Regarding Ra-226, that information could include either material-specific measurements of Ra sorption to disposal structure concrete or additional support for the revised lower inventory estimates for Ra-226 and Th-230 that DOE used in Case K. Regarding Se-79, that information could include additional model support (e.g., results of laboratory experiments) for the appropriate sorption coefficient for Se in oxidized disposal structure concrete. Alternately for Ra and Se, if the DOE dose projection changes, then NRC could determine that the potential dose from Ra and Se is appropriate without sorption in the disposal structure concrete. DOE may provide additional model support (e.g., results of laboratory experiments) to demonstrate that the sorption coefficient for Se in oxidized disposal structure concrete reflects the sorption of selenate rather than selenite. For either Ra-226 or Se-79, if appropriate information for one of those radionuclides is provided by DOE, but not the other radionuclide, then NRC could close MF 6.01 and open a new MF for the other radionuclide.

4.6.2. MF 6.02: Tc Sorption in Disposal Structure Concrete

The information in Section 4.6.2 (MF 6.02 – Tc Sorption in Disposal Structure Concrete) for §61.42 is the same as the information in Section 3.6.2 (MF 6.02 – Tc Sorption in Disposal Structure Concrete) for §61.41.

NRC expects to close MF 6.02 (Tc Sorption in Disposal Structure Concrete) under PO §61.42 after NRC determines that K_d values for Tc in reduced and oxidized disposal structure concrete are well-supported. Alternately, if the DOE dose projection changes, then NRC could determine that the potential dose from Tc is appropriate without Tc sorption in disposal structure concrete.

4.6.3. MF 6.03: Performance of Disposal Structure Roofs and HDPE/GCL Layers

The information in Section 4.6.3 (MF 6.03 – Performance of Disposal Structure Roofs and HDPE/GCL Layers) for §61.42 is the same as the information in Section 3.6.3 (MF 6.03 – Performance of Disposal Structure Roofs and HDPE/GCL Layers) for §61.41.

NRC expects to close MF 6.03 (Performance of Disposal Structure Roofs and HDPE/GCL Layers) under PO §61.42 after NRC determines that model support for the amount of water that DOE expects to be diverted by the lower lateral drainage layer, including support for the hydraulic conductivity of the relevant engineered layers, is sufficient. Alternately, NRC could close MF 6.03 if DOE conservatively assumes less diversion around the disposal structures in the PA model.

4.6.4. MF 6.04: Disposal Structure Concrete Fracturing

The information in Section 4.6.4 (MF 6.04 – Disposal Structure Concrete Fracturing) for §61.42 is the same as the information in Section 3.6.4 (MF 6.04 – Disposal Structure Concrete Fracturing) for §61.41.

NRC expects to close MF 6.04 (Disposal Structure Concrete Fracturing) under PO §61.42 after NRC determines that support for the amount of fracturing of the disposal structure floor and walls expected to occur during the performance period is adequate or if NRC determines that the estimate that DOE uses in the PA model is conservative.

4.6.5. MF 6.05: Integrity of Non-cementitious Materials

The information in Section 4.6.5 (MF 6.05 – Integrity of Non-cementitious Materials) for §61.42 is the same as the information in Section 3.6.5 (MF 6.05 – Integrity of Non-cementitious Materials) for §61.41.

NRC expects to close MF 6.05 (Integrity of Non-cementitious Materials) under PO §61.42 after NRC determines that support for the assumed performance of non-cementitious materials used in the disposal structures is adequate. For example, DOE may perform accelerated testing to estimate long-term performance. Alternately, DOE may be able to use a conservative estimate in the PA model.

4.7. MA 7 – Subsurface Transport

The modeling of the subsurface transport of radionuclides that have leached from the saltstone waste form can have a significant effect on the projected dose. The subsurface modeling generally affects the projected dose to an off-site member of the public more than the dose to an inadvertent intruder due to the longer distance to the point of compliance for the off-site member of the public. However, certain aspects of the subsurface modeling could still affect the projected dose to an inadvertent intruder. For example, the amount of sorption in the unsaturated zone beneath the disposal structures can potentially affect the projected dose to an inadvertent intruder, particularly if the release from the saltstone is pulse-like or if the radionuclide decays quickly relative to its travel time. NRC will monitor the K_d values for subsurface soil that are significant to the projected dose. For the inadvertent intruder, MF 10.09 relates specifically to model support for SRS soil K_d values for radionuclides that may become risk-significant in an updated DOE PA. That MF was included under MA 10 (Performance Assessment Model Revisions) because NRC staff expects that it will not be closed until after the PA is updated. For more information about subsurface transport, see Section 3.7 (MA 7 – Subsurface Transport).

4.7.1. MF 7.01: Certain Risk-Significant K_d Values in Site Sand and Clay

The information in Section 4.7.1 (MF 7.01 – Certain Risk-Significant K_d Values in Site Sand and Clay) for §61.42 is the same as the information in Section 3.7.1 (MF 7.01 – Certain Risk-Significant K_d Values in Site Sand and Clay) for §61.41.

NRC expects to close MF 7.01 (Certain Risk-Significant K_d Values in Site Sand and Clay) under PO §61.42 after NRC determines that site-specific measurements for the K_d value for Se in sand and clay are appropriate. Those measurements should consider the potential effect of the higher pH conditions that are likely to exist downgradient of the SDF. Alternatively, MF 7.01 may be closed if NRC determines that Se K_d values for SRS sand and clay do not have the potential to significantly affect the dose to an off-site member of the public. That determination should consider the uncertainty in other key parameters related to Se release and transport (i.e., MFs 5.04 and 6.01).

4.8. MA 8 – Environmental Monitoring

DOE conducts an effluent monitoring and environmental surveillance program on an ongoing basis at SRS. NRC staff expects that groundwater monitoring data from Z-Area will be the most appropriate to monitor in assessing compliance with §61.42 because the groundwater pathway is expected to dominate long-term doses to inadvertent intruders. Although the monitoring wells at the SDF are located further from the disposal structures than a hypothetical inadvertent intruder, groundwater data from these wells still provides useful information regarding the potential dose to an inadvertent intruder. For example, if the groundwater data indicated that an early release of radionuclides from saltstone had occurred, this could mean that the performance of the saltstone and disposal structures was worse than was assumed in the DOE PA and the dose to an intruder may be higher than was projected. The leak detectors installed beneath SDS 3A will provide important information regarding the early performance of the saltstone waste form and disposal structures. For more information about environmental monitoring, see Section 3.8 (MA 8 – Environmental Monitoring).

4.8.1. MF 8.01: Leak Detection

The information in Section 4.8.1 (MF 8.01 – Leak Detection) for §61.42 is the same as the information in Section 3.8.1 (MF 8.01 – Leak Detection) for §61.41.

NRC expects to close MF 8.01 (Leak Detection) under PO §61.42 after the leak detection system ends operation or after final waste disposal occurs, whichever comes later.

4.8.2. MF 8.02: Groundwater Monitoring

The information in Section 4.8.2 (MF 8.02 – Groundwater Monitoring) for §61.42 is the same as the information in Section 3.8.2 (MF 8.02 – Groundwater Monitoring) for §61.41.

NRC does not expect to close MF 8.02 (Groundwater Monitoring) under PO §61.42 because NRC will monitor groundwater data for the duration of NRC monitoring at the SDF.

4.9. MA 9 – Site Stability

Site stability is directly applicable to maintaining compliance with §61.44. In addition, site stability is important in both limiting infiltration through the disposal site and maintaining an adequate barrier to intrusion, where both of those are important to protection of an inadvertent intruder (§61.42). Limiting infiltration is important because NRC staff agreed with the DOE assumption that the dose to an inadvertent intruder at the SDF will be dominated by radionuclide releases to groundwater rather than direct exposure to saltstone. Site stability also enhances intruder protection because limiting erosion is important to maintaining sufficient cover depth to limit intrusion. In addition, site stability is expected to increase the probability that the site remains recognizable as an engineered facility and that the saltstone remains recognizable as waste.

4.9.1. MF 9.01: Settlement Due to Increased Overburden

The information in Section 4.9.1 (MF 9.01 – Settlement Due to Increased Overburden) for §61.42 is the same as the information in Section 3.9.1 (MF 9.01 – Settlement Due to Increased Overburden) for §61.41.

NRC expects to close MF 9.01 (Settlement Due to Increased Overburden) under PO §61.42 after NRC determines that the projections of settlement in the recent geotechnical investigations will not adversely affect SDF performance. Alternately, DOE may provide NRC information that allows NRC to determine that the new DOE settlement projections are consistent with the values assumed in the DOE 2009 PA.

4.9.2. MF 9.02: Settlement Due to Dissolution of Calcareous Sediment

The information in Section 4.9.2 (MF 9.02 – Settlement Due to Dissolution of Calcareous Sediment) for §61.42 is the same as the information in Section 3.9.2 (MF 9.02 – Settlement Due to Dissolution of Calcareous Sediment) for §61.41.

NRC expects to close MF 9.02 (Settlement Due to Dissolution of Calcareous Sediment) under PO §61.42 after NRC assesses a new DOE projection of the likelihood of the formation of sinks during the period of performance at the SDF and any resulting effects on site stability.

4.10. MA 10 – Performance Assessment Model Revisions

NRC staff identified a number of items for review when the PA is next updated under the DOE PA maintenance program. Many of the MFs identified for assessing compliance with §61.42 are identical to those identified for §61.41 because the dose to an inadvertent intruder is dominated by the groundwater dose. The main exceptions to that are the parameters related to the far-field transport. The modeling of far-field transport does not affect the dose to the inadvertent intruder because, in the DOE 2009 PA, DOE assumed that the inadvertent intruder uses water taken from directly below the disposal structure. For more information about Performance

Assessment Model Revisions, see Section 3.10 (MA 10 – Performance Assessment Model Revisions).

4.10.1. MF 10.01: Implementation of Conceptual Models

The information in Section 4.10.1 (MF 10.01 – Implementation of Conceptual Models) for §61.42 is the same as the information in Section 3.10.1 (MF 10.01 – Implementation of Conceptual Models) for §61.41.

NRC expects to close MF 10.01 (Implementation of Conceptual Models) under PO §61.42 after DOE updates the PA and NRC determines that intermediate model results are consistent with the conceptual models, quality assurance methods used are appropriate, and parameter values and uncertainty ranges are appropriate.

4.10.2. MF 10.02: Defensibility of Conceptual Models

The information in Section 4.10.2 (MF 10.02 – Defensibility of Conceptual Models) for §61.42 is the same as the information in Section 3.10.2 (MF 10.02 – Defensibility of Conceptual Models) for §61.41.

NRC expects to close MF 10.02 (Defensibility of Conceptual Models) under PO §61.42 after DOE updates the PA and NRC determines that the conceptual models are appropriate.

4.10.3. MF 10.03: Diffusivity in Degraded Saltstone

The information in Section 4.10.3 (MF 10.03 – Diffusivity in Degraded Saltstone) for §61.42 is the same as the information in Section 3.10.3 (MF 10.03 – Diffusivity in Degraded Saltstone) for §61.41.

NRC expects to close MF 10.03 (Diffusivity in Degraded Saltstone) under PO §61.42 after DOE updates the PA and NRC determines that the diffusivity information, including the model of the movement of the oxidation front, is well-supported.

4.10.4. MF 10.04: K_d Values for Saltstone

The information in Section 4.10.4 (MF 10.04 – K_d Values for Saltstone) for §61.42 is the same as the information in Section 3.10.4 (MF 10.04 – K_d Values for Saltstone) for §61.41.

NRC expects to close MF 10.04 (K_d Values for Saltstone) under PO §61.42 after DOE updates the PA and NRC determines that the K_d values for saltstone for any radionuclides that become risk-significant in the updated PA are well-supported.

4.10.5. MF 10.05: Moisture Characteristic Curves

The information in Section 4.10.5 (MF 10.05 – Moisture Characteristic Curves) for §61.42 is the same as the information in Section 3.10.5 (MF 10.05 – Moisture Characteristic Curves) for §61.41.

NRC expects to close MF 10.05 (Moisture Characteristic Curves) under PO §61.42 after DOE updates the PA and NRC determines that the MCCs are well-supported. Alternatively, MF 10.05 may be closed if, in an updated PA, DOE assumes the relative permeability is 1, which means that DOE does not use MCCs in the updated PA.

4.10.6. MF 10.06: K_d Values for Disposal Structure Concrete

The information in Section 4.10.6 (MF 10.06 – K_d Values for Disposal Structure Concrete) for §61.42 is the same as the information in Section 3.10.6 (MF 10.06 – K_d Values for Disposal Structure Concrete) for §61.41.

NRC expects to close MF 10.06 (K_d Values for Disposal Structure Concrete) under PO §61.42 after DOE updates the PA and NRC determines that the K_d values for disposal structure concrete for any radionuclides that become risk-significant in the updated PA are well-supported.

4.10.7. MF 10.07: Calculation of Build-Up in Biosphere Soil

The information in Section 4.10.7 (MF 10.07 – Calculation of Build-Up in Biosphere Soil) for §61.42 is the same as the information in Section 3.10.7 (MF 10.07 – Calculation of Build-Up in Biosphere Soil) for §61.41.

NRC expects to close MF 10.07 (Calculation of Build-Up in Biosphere Soil) under PO §61.42 after DOE updates the PA and NRC determines that the soil K_d values are well-supported in the soil build-up calculation (i.e., if DOE chose conservative low K_d values in the transport calculation, then the soil K_d values may not be the same K_d values used in the transport calculation).

4.10.8. MF 10.08: Consumption Factors and Uncertainty Distributions for Transfer Factors

The information in Section 4.10.8 (MF 10.08 – Consumption Factors and Uncertainty Distributions for Transfer Factors) for §61.42 is the same as the information in Section 3.10.8 (MF 10.07 – Consumption Factors and Uncertainty Distributions for Transfer Factors) for §61.41.

NRC expects to close MF 10.08 (Consumption Factors and Uncertainty Distributions for Transfer Factors) under PO §61.42 after DOE updates the PA and NRC determines that the values of consumption factors and uncertainty distributions for transfer factors are well-supported.

4.10.9. MF 10.09: K_d Values for SRS Soil

The projected dose to an off-site member of the public is generally more sensitive to subsurface transport of radionuclides than an inadvertent intruder. However, certain aspects of subsurface modeling could still affect the projected dose to an inadvertent intruder. NRC will monitor support for site-specific soil K_d values for radionuclides that may become risk-significant in the updated PA. For more information about K_d values for SRS Soil, see Section 3.10.9 (MF 10.09 – K_d Values for SRS Soil).

NRC expects to close MF 10.09 (K_d Values for SRS Soil) under PO §61.42 after DOE updates the PA and NRC determines that the site-specific K_d values for any radionuclides that become risk-significant in the updated PA are well-supported.

4.10.10. MF 10.10

MF 10.10 is not applicable to §61.42 (Protection of the Inadvertent Intruder) because it relates to radionuclide transport beyond the point of exposure to an intruder.

4.10.11. MF 10.11

MF 10.11 is not applicable to §61.42 (Protection of the Inadvertent Intruder) because it relates to radionuclide transport beyond the point of exposure to an intruder.

4.10.12. MF 10.12

MF 10.12 is not applicable to §61.42 (Protection of the Inadvertent Intruder) because it relates to radionuclide transport beyond the point of exposure to an intruder.

4.10.13. MF 10.13

MF 10.13 is not applicable to §61.42 (Protection of the Inadvertent Intruder) because it relates to radionuclide transport beyond the point of exposure to an intruder.

(Intentionally Left Blank)

5. MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.43

§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.*

The terms "operations" and "as low as is reasonably achievable" are defined in the Definitions section of this monitoring plan.

The NRC will continue to monitor and assess DOE compliance with the §61.43 PO through the end of the institutional control period by determining whether DOE disposal actions meet the appropriate DOE requirements, §61.41 requirements, and §20.1101(d) requirements for protection of individuals during operations. For workers performing duties on a controlled DOE site under a DOE radiation protection program, the 50 mSv/yr (5 rem/yr) radiation worker dose limit applies. 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 applies for doses from sources other than effluents. The dose to members of the public from effluents is limited to the 0.25 mSv/yr (25 mrem/yr) dose requirement of §61.41. 10 CFR 20.1101(d) further specifies a constraint of 0.10 mSv/yr (10 mrem/yr) from airborne emissions, excluding Rn-222 and its daughters, to a member of the public likely to receive the highest dose.

5.1. MA 11 – Radiation Protection Program

DOE has a radiation protection program in place to ensure the protection of individuals during operations. In the DOE 2006 final WD*,* DOE provided a crosswalk of the relevant DOE regulation or limit consistent with that in 10 CFR Part 20 to demonstrate that the DOE regulation or limit provides an equivalent level of protection as 10 CFR Part 20. In the NRC 2005 TER *(NRC, 2005)*, NRC determined that, during operations, individuals were protected by DOE regulations, which provided protection comparable to 10 CFR Part 20. Thus, NRC concluded that there was reasonable assurance that the §61.43 PO for protection of individuals during operations could be met by DOE.

During onsite observation visits in October 2007 and March 2008, NRC staff reviewed aspects of the DOE radiation protection program. As documented in the reports *(NRC 2008a, NRC 2008b)*, NRC determined that DOE had an adequate radiation protection program, including air monitoring program, in place for SDF operations.

During operations associated with salt waste disposal at the SRS, the primary effluent of concern will be air emissions because NRC staff does not expect that there will be significant releases to the subsurface or to surface water from the waste in the saltstone disposal structures during operations. Also, the release of radionuclides from the saltstone to the subsurface will be monitored to assess DOE compliance with the §61.41 and §61.42. Any leaching of contaminants from the disposal structures that is observed while the SDF is still in operation may indicate the ability of the wasteform to retain the radionuclides is worse than expected.

5.1.1. MF 11.01: Dose to Individuals During Operations

NRC staff expects compliance with the dose requirements for protection of individuals during operations to be assessed through the use of dosimetry and the monitoring of both radiation data and radiation records. At least annually, NRC staff should review DOE reports and records related to dose during waste disposal operations (e.g., records and radiological control documents associated with saltstone operations, including associated worker dose records) to evaluate whether or not the doses are within the limits in 10 CFR Part 20.

NRC staff should periodically confirm that the programs and policies in the DOE 2006 final WD continue to be in effect during the operational period. In particular, NRC staff should verify that personnel involved in the waste disposal operations are provided dosimetry and are familiar with the requirements of the DOE radiation protection program. NRC staff should verify that the dose to members of the public is assessed appropriately.

NRC expects to close MF 11.01 (Dose to Individuals During Operations) under PO §61.43 at the end of the institutional control period.

5.1.2. MF 11.02: Air Monitoring

DOE monitors the air quality at the SRS using air sampling stations located at the site boundary and in other locations throughout SRS. Also, DOE monitors the airborne effluents from operating facilities by sampling the emissions from the stacks. On an annual basis, NRC staff should review air monitoring data associated with the disposal of salt waste during the operation of the SDF to determine whether or not the activity released in the air emissions could cause a member of the public to receive an annual dose of greater than 0.10 mSv/yr (10 mrem/yr) through the air pathway.

NRC staff should periodically confirm that the DOE air monitoring program continues to adequately assess the airborne emissions from the SDF, particularly if there are any major configuration changes to either the facilities or the air sampling equipment or protocol. NRC staff should evaluate whether or not the sampling locations and sampling methodology are adequate to assess the dose to a member of the public due to airborne emissions from the SDF. NRC staff expects the dose from airborne emissions to be small. However, if the airborne emissions dose becomes more risk-significant, then NRC staff will evaluate the air monitoring program in greater detail.

NRC expects to close MF 11.02 (Air Monitoring) under PO §61.43 at the end of the institutional control period.

(Intentionally Left Blank)

6. MONITORING TO ASSESS COMPLIANCE WITH 10 CFR 61.44

§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.*

NRC will continue to assess DOE compliance with the §61.44 PO regarding closure cap integrity and protection against inadvertent intrusion. Ensuring site stability helps to minimize the access of water to the waste form by helping to maintain the performance of the closure cap. In addition, site stability is important in protecting against inadvertent intrusion because: (i), an intact erosion barrier may deter intruders; and (ii) stability increases the likelihood of the site and waste being recognizable for a long period of time.

Site stability is affected by features, events, and processes (FEPs) that affect multiple barriers and by FEPs that are more localized within a single barrier. MA 9 (Site Stability) addresses FEPs that are external to the individual disposal facility components. For example, settlement of the subsurface due to static loading and dissolution of calcareous sediment (MF 9.01 and MF 9.02) are important because settlement may affect the stability of the disposal structures, waste form, and closure cap. FEPs that are internal to individual components are discussed under the MA most related to that component. For example, erosion of the topsoil layer is discussed under MA 2 (Infiltration and Erosion Control).

6.1. MA 9 – Site Stability

The key attributes responsible for providing stability of the SDF are the grout waste form and the erosion protection barrier associated with the closure cap. The use of grout will create a solid monolith with little void space and eliminate differential settlement of the cover due to structural collapse of the disposal structures. The proper design, construction, and performance of the erosion barrier are expected to limit surface water erosion and direct contact of the waste by potential inadvertent intruders.

Site stability could also be affected by subsurface settlement, which could lead to cracking of the disposal structures and the grout waste form. Cracking is not expected to result in significant structural collapse of the disposal structures; however, the integrity of disposal structure concrete and waste form is important to radionuclide release. Settlement could impact the performance of the closure cap due to modifications of the closure cap slope and surface drainage patterns and disruption to closure cap components (e.g., HDPE/GCL composite layer, foundation layer, lateral drainage layer). The performance of the closure cap is discussed under MA 2 (Infiltration and Erosion Control).

6.1.1. MF 9.01: Settlement Due to Increased Overburden

The additional loading from the grouting of the disposal structures and the presence of the overlying closure cap will exceed preexisting conditions. That increased overburden may result in the compression of subsurface layers and, consequently, differential settlement. Differential settlement has the potential to disrupt the HDPE/GCL composite layer, which acts as a significant barrier to infiltration in the early part of the performance period. Differential settlement may also affect the performance of the foundation layer and lateral drainage layer, both of which provide long-term barriers to infiltration. In the DOE 2009 PA, DOE assumed that the settlement due to static loading will only be a few inches based on previous analyses in the F-Area and it will be uniformly distributed over the closure cap (WSRC-STI-2008-00244). However, recent geotechnical investigations at the SDF projected static settlement values that exceed the assumptions in the DOE 2009 PA. NRC will monitor the development of information related to the effects of increased overburden on site stability.

NRC expects to close MF 9.01 (Settlement Due to Increased Overburden) under PO §61.44 after NRC determines that the projections of settlement in the recent geotechnical investigations will not adversely affect SDF performance. Alternately, DOE may provide NRC information that allows NRC to determine that the new DOE settlement projections are consistent with the values assumed in the DOE 2009 PA.

6.1.2. MF 9.02: Settlement Due to Dissolution of Calcareous Sediment

Dissolution of calcareous sediment and the subsequent consolidation of soft zones may result in settlement. An ongoing evolution of the subsurface was demonstrated by evidence of dissolution based on elevated bicarbonate ion concentrations and relatively high pH values for groundwater samples collected in or near the Santee formation (WSRC-RP-92-450). Although dissolution of the calcareous sediment in the saturated zone is likely to be a very slow process, DOE has not demonstrated that dissolution is insignificant with respect to site stability over the course of a 10,000-year performance period. For example, subsidence beneath the SDF could result in fracturing in the saltstone grout and disposal structures with increased localized infiltration because of run-in from the surrounding area. NRC will monitor the development of information related to the potential for sink development. NRC staff should evaluate the DOE assumptions regarding the potential ongoing dissolution of calcareous sediment. NRC staff should evaluate any new information that DOE provides to support the determination that projected future dissolution of calcareous sediment is significant to site stability.

NRC expects to close MF 9.02 (Settlement Due to Dissolution of Calcareous Sediment) under PO §61.44 after NRC assesses a new DOE projection of the likelihood of the formation of sinks during the period of performance at the SDF and any resulting effects on site stability.

(Intentionally Left Blank)

7. REFERENCES

LWO-WSE-2009-00038, “Estimation of Variability in Premix Preparation at Saltstone.” Savannah River Site. Aiken, SC. February 2009.

Pabalan, R.T., Alexander, G.W., and D.J. Waiting. “Experimental Study of Contaminant Release from Reducing Grout.” Center for Nuclear Waste Regulatory Analyses. San Antonio, TX. March 2012. ML12089A319.

Savannah River National Laboratory (SRNL). SRNL-ESB-2008-00017, “Video Survey of Saltstone Fault 4, Cell G.” Savannah River Site. Aiken, SC. April 2008. ML113320354.

_____. “Photos Taken During February 8-11 2010, Saltstone Observation Visit.” Savannah River Site. Aiken, SC. February 2009. ML100550095.

———. SRNL-STI-2009-00419, “Hydraulic and Physical Properties of ARP/MCU Saltstone Grout.” Savannah River Site. Aiken, SC. May 2010. ML102160544.

_____. SRNL-STI-2009-00804, “Characterization of Core Sample Collected from the Saltstone Disposal Facility.” Savannah River Site. Aiken, SC. December 2009. ML102180278.

———. SRNL-STI-2010-00167, “Design and Testing of the Formed-Core Sampling System for Saltstone Facility Vaults.” Savannah River Site. Aiken, SC. March 2010. ML102180289.

_____. SRNL-STI-2010-00657, “Permeability Testing of Simulated Saltstone Core Sample Collected and Vault 4 Cell E Saltstone.” Savannah River Site. Aiken, SC. October 2010.

_____. SRNL-STI-2010-00667, “Distribution Coefficients (K_d) Generated from a Core Sample Collected from the Saltstone Disposal Facility.” Savannah River Site. Aiken, SC. April 2011. ML113320284.

_____. SRNL-STI-2010-00668, “Long-term Technetium Interactions with Reducing Cementitious Materials.” Savannah River Site. Aiken, SC. March 2011. ML111290356.

———. SRNL-STI-2011-00665, “Operational and Compositional Factors that Affect the Performance Properties of ARP/MCU Saltstone Grout.” Savannah River Site. Aiken, SC. May 2011.

Savannah River Nuclear Solutions. K-ESR-Z-00002 Rev. 0, “Saltstone Disposal Cells No. 3 and 5 Geotechnical Investigation Report.” Savannah River Site. Aiken, SC. July 2009. ML102160523.

———. SRNS-STI-2008-00045, “Saltstone and Concrete Interactions with Radionuclides: Sorption (K_d), Desorption, and Reduction Capacity Measurements.” Savannah River National Laboratory. Aiken, SC. October 2008. ML090150234.

Savannah River Remediation (SRR). SRR-CWDA-2009-00017, “Performance Assessment for the Saltstone Disposal Facility at the Savannah River Site.” Savannah River Site. Aiken, SC. October 2009. ML101590008.

_____. SRR-CWDA-2009-00054, “Comment Response Matrix for NRC Comments on the F-Tank Farm Performance Assessment (SRS-REG-2007-00002, Rev. 0, dated June 27, 2008).” Savannah River Site. Aiken, SC. December 2009. ML100550169.

_____. SRR-CWDA-2010-00033, “Comment Response Matrix for NRC – Requests for Additional Information on the Saltstone Disposal Facility Performance Assessment.” Savannah River Site. Aiken, SC. July 2010. ML102090664.

_____. SRR-CWDA-2010-00043, “DOE Incoming NRC Onsite Observation [April 19, 2010] Disposal Unit 2 Tour Photos Presentation.” Savannah River Site. Aiken, SC. July 2010. ML101900021.

_____. SRR-CWDA-2011-00044, Rev. 1, “Comment Response Matrix for NRC – Second Request for Additional Information on the Saltstone Disposal Facility Performance Assessment.” Savannah River Site. Aiken, SC. August 2011. ML113320303.

_____. SRR-CWDA-2011-00054, Rev. 1, “Comment Response Matrix for United States Nuclear Regulatory Commission Staff Comments on the Draft Basis for Section 3116 Determination and Associated Performance Assessment for the F-Tank Farm at the Savannah River Site,” Savannah River Site. Aiken, SC. June 2011

_____. SRR-CWDA-2011-00105, “Saltstone Disposal Facility Vault 4 Video Inspection Summary.” Savannah River Site. Aiken, SC. July 2011.

_____. SRR-CWDA-2012-00002, “Revised Methodology for Determination of Inventories in SDF Vaults 1 and 4 through 9/30/2011.” Savannah River Site. Aiken, SC. January 2012. ML12171A395.

_____. SRR-CWDA-2013-00049, Rev. 1, “FY 2013 Implementation Plan for Savannah River Site Liquid Waste Facilities Performance Assessment Maintenance Program.” Savannah River Site. Aiken, SC. 2013.

_____. SRR-TR-2010-00374, “Z-Area Groundwater Monitoring Report for 2010.” Savannah River Site. Aiken, SC. December 2010.

South Carolina. 07-ALJ-07-121-CC, “NRDCE, et al vs. SCDHEC, et al.” South Carolina Administrative Law Court. South Carolina. 2007.

Tallent, O.K., McDaniel, E.W., Del Cul, G.D., Dodson, K.E., and D.R. Trotter, CONF-871124-2, “Immobilization of Technetium and Nitrate in Cement-Based Materials.” Oak Ridge National Laboratory. Oak Ridge, TN. 1987.

Thayer, P.A., Harris, M.K. and Blount, G.C., “Case Studies of the Effect of Subsurface Pseudo-Karst on Preferential Contaminant Transport at the Savannah River Site, South Carolina, USA.” Presented at the 5th International Symposium and Field Seminar on Karst Water and Environmental Impacts (TURKEY95).” Antalya, Turkey. September 1995.

T-CLC-Z-00006, Rev. 0, “Saltstone Vault Structural Degradation Prediction.” Savannah River Site. Aiken, SC. July 2003. ML102210070.

U.S. Department of Energy (DOE). DOE Manual 435.1-1, Change 1, “Radioactive Waste Management Manual.” DOE. Washington, DC. June 2001. ML110800193.

_____. DOE Order 435.1. "Radioactive Waste Management." DOE, Washington, DC. August 2001. ML101590125.

_____. DOE-WD-2005-001, Rev. 1, "Basis for Section 3116 Determination Salt Waste Disposal at the Savannah River Site." DOE, Washington, DC. January 2006. ML102850319.

U.S. Nuclear Regulatory Commission (NRC). NUREG-0782, "Draft Environmental Impact Statement on 10 CFR Part 61 'Licensing Requirements for Land Disposal of Radioactive Waste'." NRC. Washington, DC. September, 1981. ML052590348. *(NRC, 1981)*

_____. NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." NRC. Washington, DC. November 1996. ML033500190. *(NRC, 1996)*

_____. SRM-SECY-01-0148, "Processes for Revision of 10 CFR Part 20 Regarding Adoption of ICRP Recommendations on Occupational Dose Limits and Dosimetric Models and Parameters." NRC. Washington, DC. April 2001. ML021050104. *(NRC, 2001)*

_____. NUREG-1623, "Design of Erosion Protection for Long-Term Stabilization."   NRC. Washington, DC. September 2002. ML022530043. *(NRC, 2002)*

_____. "NRC Technical Evaluation Report for the U.S. DOE Savannah River Site Draft Section 3116 Waste Determination for Salt Waste Disposal." NRC. Washington, DC. December 2005. ML053010225. *(NRC, 2005)*

_____. "NRC Plan for Monitoring the U.S. DOE Salt Waste Disposal at the Savannah River Site in Accordance with the National Defense Authorization Act for Fiscal Year 2005." NRC. Washington, DC. May 2007. ML070730363. *(NRC, 2007a)*

_____. NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. DOE Waste Determinations – Draft Final Report for Interim Use." NRC. Washington, DC. August 2007. ML072360184. *(NRC, 2007b)*

_____. "NRC [October 29-30, 2007,] Onsite Observation Report for the Savannah River Site Saltstone Production and Disposal Facilities." NRC. Washington, DC. January 2008. ML073461038. *(NRC, 2008a)*

_____. "NRC [March 24-28, 2008,] Onsite Observation Report for the Savannah River Site Saltstone Production and Disposal Facilities." NRC. Washington, DC. June 2008. ML081290367. *(NRC, 2008b)*

_____. "Memorandum to Gregory Suber – Technical Review: Thermodynamic and Mass Balance Analysis of Expansive Phase Precipitation in Saltstone." NRC. Washington, DC. November 2008. ML093030220. *(NRC, 2008c)*

_____. "NRC Staff Comments on the Performance Assessment for the F-Tank Farm at the Savannah Rivers Site." NRC. Washington, DC. January 2009. ML1090150222. *(NRC, 2009a)*

_____. "NRC March 25-26, 2009, Onsite Observation Report for the Savannah River Site Saltstone Facility." NRC. Washington, DC. May 2009. ML091320439. *(NRC, 2009b)*

_____. “NRC June 3, 2009, Onsite Observation Report for the Savannah River Site Saltstone Facility.” NRC. Washington, DC. September 2009. ML092170006. *(NRC, 2009c)*

_____. “NRC February 9-11, 2010, Onsite Observation Report for the Savannah River Site Saltstone Facility.” NRC. Washington, DC. June 2010. ML101320386. *(NRC, 2010a)*

_____. “NRC April 19, 2010, Onsite Observation Report for the Savannah River Site Saltstone Facility.” NRC. Washington, DC. July 2010. ML101660451. *(NRC, 2010b)*

_____. “NRC July 28, 2010, Onsite Observation Report for the Savannah River Site Saltstone Facility.” NRC. Washington, DC. November 2010. ML102180250. *(NRC, 2010c)*

_____. “Second Request for Additional Information on the 2009 Performance Assessment for the Saltstone Disposal Facility at the Savannah River Site.” NRC. Washington, DC. December 2010. ML103400571. *(NRC, 2010d)*

_____. “NRC January 26, 2011, Onsite Observation Report for the Savannah River Site Saltstone Facility.” NRC. Washington, DC. March 2011. ML110590941. *(NRC, 2011a)*

_____. “NRC April 26, 2011, Onsite Observation Report for the Savannah River Site Saltstone Facility.” NRC. Washington, DC. April 2011. ML111890319. *(NRC, 2011b)*

_____. “Technical Evaluation Report for the Revised Performance Assessment for the Saltstone Disposal Facility at the Savannah River Site, South Carolina.” NRC. Washington, DC. April 2012. ML121020140. *(NRC, 2012)*

Washington Savannah River Company, K-ESR-Z-00001, Rev. 0, “Saltstone Vault No. 2 Geotechnical Investigation Report. Savannah River Site.” Aiken, SC. 2006. ML101590273.

Westinghouse Savannah River Company. WSRC-RP-92-450, “Groundwater Geochemistry of the Savannah River Site and Vicinity.” Savannah River Site. Aiken, SC. 1992. ML111240587.

_____. WSRC-STI-2006-0000037, “Influence of Dissolved Organic Carbon and pH on Iodine, Perrhenate, and Selenate Sorption to Sediment.” Savannah River Site. Aiken, SC. September 2006. ML102160592.

_____. WSRC-STI-2008-00236, “Thermodynamic and Mass Balance Analysis of Expansive Phase Precipitation in Saltstone.” Savannah River Site. Aiken, SC. May 2008. ML083400055.

_____. WSRC-STI-2008-00244, “Saltstone Disposal Facility Closure Cap Concept and Infiltration Estimates.” Savannah River Site. Aiken, SC. May 2008. ML101600430.

_____. WSRC-TR-2005-00257, Rev. 4, “Groundwater Monitoring Plan for the Z-Area Saltstone Disposal Facility.” Savannah River Site. Aiken, SC. September 2006.

_____. WSRC-TR-2008-00001, “Z-Area Groundwater Monitoring Report for 2007.” Savannah River Site. Aiken, SC. January 2008. ML111290392.

X-CLC-Z-00027, “Inventory Determination of PODD Radionuclides in Saltstone Vaults 1 and 4.” June 2009. ML102160640.

X-CLC-Z-00034, “Saltstone Observation Documents – Inventory Determination of PODD-SA Radionuclides in the Saltstone Disposal Facility through 09/30/10.” December 2010 ML111310276.

8. LIST OF CONTRIBUTORS

U.S. Nuclear Regulatory Commission	
Name	**Title/Degrees**
Harry Felsher	Senior Project Manager M.S., Nuclear Engineering, The Ohio State University M.S., Nuclear Engineering, Texas A&M University B.S., Engineering, University of Maryland
Nishka Devaser	Project Manager M.S., Nuclear Engineering, University of Tennessee B.S., Nuclear Engineering, University of Tennessee
A. Christianne Ridge	Senior Systems Performance Analyst Ph.D., Environmental Engineering, University of California, Berkeley M.S., Environmental Engineering, Cornell University B.A., Physics, Drew University
Karen E. Pinkston	Systems Performance Analyst Ph.D., Environmental Engineering, University of California, Berkeley M.S., Environmental Engineering, University of California, Berkeley B.S., Civil Engineering, Carnegie Mellon University
George Alexander	Systems Performance Analyst Ph.D., Energy & Geo-Environmental Engineering, The Pennsylvania State University M.S., Energy & Geo-Environmental Engineering, The Pennsylvania State University B.S., Geo-Environmental Engineering, The Pennsylvania State University

APPENDIX A: TECHNICAL NOTES FOR MONITORING FACTORS

(Intentionally Left Blank)

A.1 Monitoring Area 1 – Inventory

Technical Notes for MF 1.01: Inventory in Disposal Structures

When monitoring the inventory in disposal structures at the SDF, NRC staff should track the inventory that is disposed in the individual disposal structures at the SDF. The inventory should be compared to the inventory provided in the DOE analyses (see Table A-1). If the inventory of any radionuclide in any disposal structure is greater than the inventory in Table A-1, then NRC staff should perform an assessment of the dose implications to both an off-site member of the public and an inadvertent intruder. For radionuclides that were not originally projected to be significant to dose, a simple calculation may be sufficient to demonstrate that the dose from the radionuclide remains low. In performing the assessment of the dose implications, it is important to consider both the inventory in the disposal structure in question as well as the inventory in neighboring disposal structures. This is important because if the inventory in nearby disposal structures is lower than was assumed in the PA, then the overall effect of the cumulative dose may not be significant. Alternatively, if adjacent disposal structures also have inventory that exceeds the Table A-1 inventory, then the total dose may be higher than was projected in the PA and associated DOE analyses.

Information on the inventory disposed at the SDF on a radionuclide basis can be found in annual inventory reports that summarize the current inventory in the disposal structures at the SDF (e.g., X-CLC-Z-00027, X-CLC-Z-00034, and SRR-CWDA-2012-00002). Additional quarterly reports containing information on the volume of salt waste disposed, volume of saltstone produced, and concentration of select radionuclides in the salt waste disposed in the quarter are posted on the SRS website (http://sro.srs.gov/saltstone.htm). The quarterly reports are initially published with estimated information and later are revised with updated values after key information, such as the results of waste samples, becomes available. NRC staff can use the volume and concentration information in the quarterly reports to estimate the inventory disposed during the year.

Table A-1: Projected Inventory at Time of Closure

Isotope	SDS 1 (Ci) **	SDS 4 (Ci) **	Each Disposal Structure Similar to SDS 2A (64 total) (Ci) **	Total SDF (Ci)
Ac-227		1.60×10^{-5}	1.70×10^{-7}	2.70×10^{-5}
Al-26		3.40×10^{-1}	1.90×10^{-1}	1.30×10^{1}
Am-241	4.70×10^{-4}	1.30×10^{2}	1.40	2.20×10^{2}
Am-242m		6.70×10^{-2}	5.90×10^{-4}	1.00×10^{-1}
Am-243		1.80	3.70×10^{-2}	4.20
Ba-137m	4.10	2.80×10^{5}	2.20×10^{1}	2.80×10^{5}
Bk-249		1.80×10^{-28}	1.80×10^{-28}	1.20×10^{-26}
C-14	1.30	2.70×10^{1}	2.00	1.60×10^{2}
Ce-144		1.80×10^{-9}	3.60×10^{-10}	2.50×10^{-8}
Cf-249		6.50×10^{-13}	6.70×10^{-13}	4.40×10^{-11}
Cf-251		1.20	2.30×10^{-14}	1.20
Cf-252		1.80×10^{-18}	1.80×10^{-18}	1.20×10^{-16}
Cl-36	7.60×10^{-4}	3.00×10^{-3}	4.20×10^{-4}	3.10×10^{-2}

Isotope	SDS 1 (Ci) **	SDS 4 (Ci) **	Each Disposal Structure Similar to SDS 2A (64 total) (Ci) **	Total SDF (Ci)
Cm-242		6.70×10^{-2}	6.30×10^{-19}	6.70×10^{-2}
Cm-243		2.10×10^{-1}	2.10×10^{-4}	2.20×10^{-1}
Cm-244		1.30×10^{2}	9.50×10^{-1}	1.90×10^{2}
Cm-245		9.20×10^{-1}	2.40×10^{-4}	9.40×10^{-1}
Cm-247		3.90×10^{-6}	7.10×10^{-14}	3.90×10^{-6}
Cm-248		1.20×10^{-13}	7.40×10^{-14}	4.90×10^{-12}
Co-60	8.20×10^{-5}	4.60×10^{-1}	5.40×10^{-2}	3.90
Cs-134		5.20×10^{-1}	1.50×10^{-5}	5.20×10^{-1}
Cs-135		5.40	1.30×10^{-4}	5.40
Cs-137	4.30	3.00×10^{5}	2.30×10^{1}	3.00×10^{5}
Eu-152	1.80×10^{-3}	9.70×10^{-2}	9.80×10^{-2}	6.40
Eu-154	2.30×10^{-4}	1.20×10^{1}	1.80	1.30×10^{2}
Eu-155		6.80×10^{-1}	1.30×10^{-1}	9.00
H-3	6.10	2.60×10^{2}	3.00×10^{1}	2.20×10^{3}
I-129	1.10×10^{-1}	2.80×10^{-1}	3.80×10^{-1}	2.50×10^{1}
K-40	7.60×10^{-4}	3.00×10^{-3}	4.20×10^{-4}	3.10×10^{-2}
Na-22		1.50×10^{-1}	6.90×10^{-2}	4.60
Nb-93m	2.50×10^{-1}	8.40	3.70×10^{-1}	3.20×10^{1}
Nb-94	2.50×10^{-3}	8.70×10^{-2}	3.80×10^{-3}	3.30×10^{-1}
Ni-59	3.50×10^{-2}	4.00×10^{-1}	8.40×10^{-2}	5.80
Ni-63	7.80×10^{-1}	2.20×10^{1}	2.40	1.80×10^{2}
Np-237	4.50×10^{-3}	6.10×10^{-1}	5.00×10^{-2}	3.80
Pa-231		9.30×10^{-5}	9.80×10^{-7}	1.60×10^{-4}
Pd-107	1.90×10^{-3}	5.00×10^{-2}	5.60×10^{-3}	4.10×10^{-1}
Pm-147		4.10×10^{-1}	7.70×10^{-2}	5.30
Pr-144		1.80×10^{-9}	3.60×10^{-10}	2.50×10^{-8}
Pt-193	3.70×10^{-1}	1.00×10^{1}	1.10	8.10×10^{1}
Pu-238	7.80×10^{-3}	1.0×10^{3} *	1.70×10^{2} *	1.19×10^{4}
Pu-239	1.20×10^{-2}	3.80×10^{2}	1.50×10^{1}	1.30×10^{3}
Pu-240	1.20×10^{-2}	1.20×10^{2}	4.10	3.80×10^{2}
Pu-241	9.80×10^{-3}	2.40×10^{3}	4.20×10^{1}	5.10×10^{3}
Pu-242	9.00×10^{-4}	8.10×10^{-1}	3.90×10^{-3}	1.10
Pu-244		1.60×10^{-2}	1.60×10^{-5}	1.70×10^{-2}
Ra-226	6.40×10^{-7}	1.0×10^{-3} *	1.3×10^{-5} *	1.83×10^{-3}
Ra-228		1.60×10^{-6}	8.70×10^{-5}	5.60×10^{-3}
Rh-106	1.50×10^{-10}	9.10×10^{-7}	1.20×10^{-6}	7.80×10^{-5}
Ru-106	1.50×10^{-10}	9.10×10^{-7}	1.20×10^{-6}	7.80×10^{-5}
Sb-125	1.60×10^{-1}	5.70	2.40×10^{-1}	2.10×10^{1}
Sb-126	1.40×10^{-1}	9.00×10^{-1}	1.20	7.80×10^{1}
Sb-126m	1.00	6.40	8.20	5.30×10^{2}
Se-79	3.00×10^{-1}	4.60×10^{1}	1.40	1.40×10^{2}

Isotope	SDS 1 (Ci) **	SDS 4 (Ci) **	Each Disposal Structure Similar to SDS 2A (64 total) (Ci) **	Total SDF (Ci)
Sm-151		4.20×10^{1}	5.90×10^{1}	3.80×10^{3}
Sn-126	1.00	6.40	8.20	5.30×10^{2}
Sr-90	6.90×10^{-3}	2.40×10^{5}	3.70×10^{1}	2.40×10^{5}
Tc-99	1.10×10^{2}	5.80×10^{2}	5.40×10^{2}	3.50×10^{4}
Te-125m	3.80×10^{-2}	1.40	5.80×10^{-2}	5.20
Th-229	3.00×10^{-1}	2.50×10^{1}	3.90×10^{-2}	2.80×10^{1}
Th-230	4.10×10^{-1}	1.0×10^{-2} *	1.3×10^{-4} *	4.28×10^{-1}
Th-232		3.20×10^{-4}	1.40×10^{-3}	9.00×10^{-2}
U-232		4.40×10^{-2}	3.10×10^{-4}	6.40×10^{-2}
U-233	2.80×10^{-1}	2.40×10^{1}	3.70×10^{-2}	2.70×10^{1}
U-234	2.80×10^{-1}	1.0×10^{1} *	1.30×10^{-1} *	1.86×10^{1}
U-235	3.20×10^{-3}	4.70×10^{-1}	3.00×10^{-3}	6.70×10^{-1}
U-236	3.20×10^{-3}	7.70×10^{-1}	1.60×10^{-2}	1.80
U-238	7.40×10^{-3}	5.90×10^{-1}	1.00×10^{-1}	7.00
Y-90	6.90×10^{-3}	2.40×10^{5}	3.70×10^{1}	2.40×10^{5}
Zr-93	2.50×10^{-1}	8.40	3.70×10^{-1}	3.20×10^{1}
Total	**1.30×10^{2}**	**1.10×10^{6}**	**1.00×10^{3}**	**1.10×10^{6}**
1.0 MBq is 3.7×10^{4} Ci				
from DOE 2009 PA; Tables 3.3-1, 3.3-3, 3.3-5, and 3.3-7 *except * from SRR-CWDA-2011-00044; Response to PA-8.*				
*** DOE refers to SDS 1 as Vault 1, SDS 4 as Vault 4, SDS 2A as an FDC, and other disposal structures similar to SDS 2A as FDCs*				

<u>Technical Notes for MF 1.02: Methods Used to Assess Inventory</u>

When monitoring the methods used to assess the inventory, NRC staff should evaluate the DOE approach for sampling the waste, the measurement of radionuclides in the waste samples, the methods used to estimate the concentration of radionuclides present at levels below the detection limit, and the methods used to track the inventory in the disposal structures. NRC staff should focus on radionuclides that are currently identified as risk-significant as well as relevant ancestors (e.g., Tc-99, Ra-226, [Th-230,] I-129), but will also consider radionuclides that could become more risk significant if the inventory increases significantly or if modeling assumptions change (e.g., Se-79, Sr-90).

Samples of salt waste are obtained quarterly from Tank 50 (i.e., feed tank for the SDF). In reviewing the methodology for obtaining waste samples, NRC staff should consider the sampling frequency and whether the samples are representative. For example, NRC staff should consider whether the waste that was sampled is representative of the waste disposed during the quarter. If significant changes to the waste stream are expected to occur, then the sample frequency may need to be increased to capture these changes. NRC staff should consider if the samples appropriately represent the waste that is in Tank 50. For example, particles in Tank 50 may be present in higher concentrations at the bottom of the tank due to settling and, because those sludge particles tend to have a high inventory, samples taken from the top of the tank may underestimate the inventory in the waste. NRC staff previously reviewed the procedures DOE used for obtaining samples during an onsite observation visit

(NRC, 2008b) and determined that this methodology appeared to be appropriate. NRC staff should continue to evaluate the sampling methodology as significant changes to the salt waste disposal process occur (e.g., Salt Waste Processing Facility coming online).

NRC staff should review the analytical methods used in quantifying the constituents in the samples. NRC staff previously evaluated this during an onsite observation visit *(NRC, 2008b)* and determined that those methods appeared to be appropriate. The current determination of the inventory for Ra-226 and Th-230 are limited by their analytical detection limits and, because those radionuclides are potentially important to dose, NRC staff should evaluate if improvements to the detection limit are possible for these radionuclides.

NRC staff should review the methods used to estimate the concentration of radionuclides present at levels below the detection limit. The DOE Revised Methodology for Determination of Inventories in SDS 1 and SDS 4 through September 30, 2011 (SRR-CWDA-2012-00002), states: "This assessment of inventory relies, whenever possible, on sample detection information from the feed material, salt solution in Tank 50. Due to the limitations of analytical methods and the large number of reported radionuclides, many of the values reported in inventory calculations are based on detection limits, special methods or assumed values." NRC staff agrees that the inventory should be based on sample results whenever possible. The reliance on process knowledge, special methods, or assumptions in establishing the inventory can lead to large uncertainties in the inventory and, consequently, large uncertainties in the dose. NRC staff should review in detail the inventory generated using these methods for radionuclides that are potentially risk-significant. In particular, NRC staff should carefully review the methodology used to develop the inventories of Ra-226 and Th-230.

Finally, NRC staff should review the methods used by DOE to track the inventory that has been disposed at the SDF. NRC staff has reviewed DOE inventory tracking methods during previous onsite observation visits *(NRC, 2008a; NRC, 2008b; NRC, 2009b; NRC, 2010a; and (NRC, 2011b)*. NRC staff should continue to evaluate this aspect of the inventory determination as waste is disposed at the SDF.

A.2 Monitoring Area 2 – Infiltration and Erosion Control

<u>Technical Notes for MF 2.01: Hydraulic Performance of Closure Cap</u>

In the 2009 PA, DOE projected that the sensitivity of the SDF performance to the closure cap would be limited. That result was largely due to the modeled shedding of water around the disposal structures (see NRC 2012 TER, Section 2.13.3.2). If the modeled diversion of water around the disposal structures is determined to be optimistic, then the performance of the closure cap would become more risk-significant. Accordingly, NRC staff should review information relevant to the assumed performance of the closure cap. Infiltration through the closure cap is controlled initially after site closure by the HDPE/GCL composite layer in the closure cap (i.e., not the composite layer on top of the disposal structures) and by the foundation layer throughout the remainder of the performance period.

Although the HDPE/GCL composite layer in the closure cap does not contribute to the long-term performance of the closure cap, DOE assumes it to be a significant barrier to infiltration in the first several hundred years after site closure (WSRC-STI-2008-00244). The performance of the composite layer is sensitive to construction quality and differential settlement (see Section 6.1.1 of this document). Accordingly, NRC staff should evaluate the QA/QC for closure cap construction and review relevant studies and testing related to HDPE/GCL performance, including testing of the integrity of the seams.

DOE modeling indicated that saturated conditions will occur above the upper composite layer in the closure cap. DOE projects an initial hydraulic head of 9.55 cm (3.76 in) will develop on top of the HDPE geomembrane and increase until 5,400 years after closure, when it is projected to range from 99.0 to 100 cm [39.0 to 40.0 in] until 10,000 years or more after closure (SRR-CWDA-2011-00044). In the NRC second Request for Additional Information during development of the NRC 2012 TER, NRC staff identified the concern that pore-pressure build-up in the overlying closure cap layers could affect cover stability, vegetation, hydraulic performance of cover materials, and erosion *(NRC, 2011b)*. DOE indicated that conservative modeling assumptions (e.g., depth of the evapotranspiration zone and degradation of the lateral drainage layer) resulted in estimates of head on the HDPE geomembrane that are bounding and conservative. Should the buildup of hydraulic head occur, DOE does not believe it would adversely impact the physical stability of the closure cap, vegetation, erosion, or performance of the composite layer. NRC staff should review information related to the likelihood of hydraulic head buildup within the cover, which may include a more realistic representation of infiltration and saturation within the proposed closure cap. If an analysis containing a more realistic representation indicates that the buildup of hydraulic head is realistic, then NRC staff should evaluate the development of information to support the projected performance of the closure cap layers under saturated conditions.

The hydraulic properties of the foundation layer should not have a significant effect on infiltration until the overlying geomembrane and geosynthetic clay liner have degraded to the point where significant water can reach the foundation layer. DOE projected that this could occur after approximately 2,500 years. DOE assumed that the foundation layer would have a saturated hydraulic conductivity of 1.0×10^{-6} cm/s [3.3×10^{-8} ft/s]. That hydraulic conductivity would limit the maximum modeled infiltration rate to approximately 32 cm/yr [12 in/yr], which is slightly less than the estimated background infiltration rate of 38 cm/yr [15 in/yr]. That constraint results in a long-term infiltration rate that is less than natural infiltration, which is implemented in the near-field PORFLOW model. A more permeable foundation layer would presumably allow more infiltration at higher annual precipitation and would increase the calculated long-term infiltration average.

Technical Notes for MF 2.02: Erosion Protection

The erosion barrier is important in maintaining the physical stability of the closure cap, which protects the waste from exposure due to erosion, and maintaining adequate cover depth to discourage inadvertent intrusion. When NRC staff reviewed a similar closure cap design for the FTF, NRC staff recommended that a preliminary evaluation of rock sources be conducted to provide confidence that an acceptable rock source is available *(NRC, 2009a)*. In response, DOE indicated that the design information was sufficient for planning purposes and that rock sources will be evaluated in the final closure cap design (SRR-CWDA-2009-00054). NRC staff remains concerned that if a rock source is not available that can adequately resist weathering, then modifications to the closure cap and/or assumptions regarding performance of the closure cap may be needed. Those modifications may be more easily accommodated earlier in the closure process. Accordingly, NRC staff should review information related to the erosion barrier to verify that a rock source is available that is capable of resisting the anticipated weathering throughout the performance period.

Erosion of the upper layers of the cap (i.e., above the erosion barrier) also could degrade other aspects of cap performance. Specifically, long-term maintenance of the topsoil and vegetative cover is important to closure cap performance because the average evapotranspiration rate dominates the modeled water balance distribution for SRS precipitation. DOE estimated the rate of erosion of the vegetative and topsoil layers using the Universal Soil Loss Equation assuming a mixed Bahia grass and pine tree cover during the post-institutional control period. The resistance of the topsoil portion of the cover to gully erosion was evaluated using the methodology in NUREG-1623 based on a Probable Maximum Precipitation (PMP) event assuming Bahia grass cover. However, DOE has not evaluated the potential cumulative effects from less significant, but more frequent precipitation events on gully formation over long time periods. NRC staff should verify the assumption that erosion based on the PMP for the vegetative and topsoil layers is conservative. In addition, NRC staff should evaluate the stability of a degraded vegetative cover because the Bahia grass, bamboo, or pine forest could be degraded by fire or extended drought, which could adversely affect the ability of the vegetative and topsoil layers to resist erosion.

A.3 Monitoring Area 3 – Waste Form Hydraulic Performance

Technical Notes for MF 3.01: Hydraulic Conductivity of Field-Emplaced Saltstone

The values of saltstone hydraulic conductivity DOE used in Case A of the DOE 2009 PA were based on measurements of laboratory-made samples. It is unclear to NRC staff whether the laboratory samples reflect the composition (e.g., actual water-to-cement ratios) and curing conditions (e.g., actual curing temperatures and humidity) of field-emplaced saltstone. In addition, it is unclear to NRC staff whether laboratory samples adequately capture the effects of scale (e.g., effects of fractures or interfaces between lifts).

Several measurements of the hydraulic conductivity of saltstone samples taken from SDS 4, Cell E (SRNL-STI-2010-00657) exceeded the 2.0×10^{-9} cm/s saltstone hydraulic conductivity of Case A of the DOE 2009 PA. DOE had suggested that the core sample measurements were artificially increased by damage sustained during coring. However, a subsequent report by Savannah River National Laboratory (SRNL) showed that laboratory-made samples subject to the same coring technique had measured hydraulic conductivities similar to laboratory-cast samples that were not subject to coring (SRNL-STI-2010-00657). That result led NRC staff to question whether the elevated hydraulic conductivity measurements of the cored samples were attributable to the coring technique or instead reflected differences between laboratory-made and field-emplaced samples (see NRC 2012 TER, Section 2.6). DOE subsequently suggested that the same coring technique yielded different results in the laboratory and the field because the coring technique was more difficult to apply in the field.

In response to the difficulty of applying the coring technique in the field, DOE developed a formed-core technique to sample field-emplaced saltstone (SRNL-STI-2010-00167). NRC staff discussed the formed-core technique with DOE and suggested that it may be difficult to adequately represent the boundary conditions of field-emplaced saltstone *(NRC, 2010c)*. In addition, it is not clear to NRC staff how DOE will account for scale effects that may not be captured with any small-scale core sample (e.g., fractures may not be present in individual small samples). If DOE uses the formed-core sampling technique to sample field-emplaced saltstone, then NRC staff should evaluate the measured hydraulic conductivity of saltstone samples collected with the formed-core technique. NRC staff should assess whether the samples adequately represent field-emplaced saltstone. NRC staff should evaluate the appropriateness of any new sampling technique. In addition, NRC staff should consider hydraulic conductivity values from any other representative studies.

Technical Notes for MF 3.02: Variability of Field-Emplaced Saltstone

The potential variability of saltstone properties may be determined directly from sampling field-emplaced saltstone and testing hydraulic properties or indirectly through laboratory experiments (see Technical Notes for MF 3.03). Laboratory experiments designed to test saltstone variability would test the effects of varying key parameters in ranges observed in field-emplaced saltstone (e.g., curing temperatures, water-to-cementitious material ratios, aluminate concentrations, presence of admixtures). In SRNL-STI-2009-00419, DOE studied the hydraulic and sorptive properties of laboratory-prepared saltstone samples that were prepared with different compositions and curing temperatures. That study did not show an effect of admixtures, organic content, or aluminate concentration; however, NRC staff determined that the small number of samples used may have limited the study's ability to detect the effects. Despite the small number of samples used, that study did indicate statistically significant effects of the salt waste to premix ratio and curing temperature on saltstone hydraulic conductivity.

DOE concluded that there is relatively little variability in the dry bulk components of saltstone (LWO-WSE-2009-00038). NRC staff analyses of the results of the DOE data in that document showed a small variation in the contributions of fly ash and slag with a larger variation in the relative mass of cement used (i.e., relative standard deviation of ±9%) (see NRC 2012 TER, Section 2.6). As NRC staff discussed with DOE during an onsite observation visit *(NRC 2008a)*, actual water-to-cement ratios may vary in part because of the addition of delivery-line flush water to the disposal structures. In addition, aluminate concentrations, which affect saltstone properties (i.e., WSRC-STI-2007-00506; SRNL-STI-2009-00184; SRNL-STI-2009-00546; SRNL-STI-2009-00810; SRNL-STI-2011-00665) depend on the salt solution and therefore may vary from batch to batch. Similarly, as discussed in SNRL-STI-2010-00522, DOE adds tributylphosphate as an antifoam agent, which may increase the solubility of certain radionuclides. NRC staff is concerned that these variations, or other changes in saltstone composition or curing conditions, may affect the hydraulic properties of saltstone grout.

Curing temperature appears to have a significant effect on saltstone hydraulic properties. Thus, variations in curing temperature are expected to be an important source of variability in the properties of field-emplaced saltstone. In coordination with NRC staff activities related to MF 3.04, NRC staff should use data from thermocouples embedded in saltstone to evaluate the potential variability in saltstone properties caused by the curing temperature profiles of field-emplaced saltstone.

Parameters related to the composition, placement, and curing of saltstone should be limited to ranges in which the hydraulic properties of the resulting waste form are well supported. NRC staff should pay particular attention to any changes in the saltstone formula and any additions not considered in laboratory experiments (e.g. flush water or chemical additions made in the field to improve processability). NRC staff should evaluate the DOE quality control program for saltstone production and the DOE processes for controlling factors important to saltstone hydraulic properties.

Technical Notes for MF 3.03: Applicability of Laboratory Data to Field-Emplaced Saltstone

The applicability of data based on laboratory-made samples to field-emplaced saltstone is important if DOE uses the properties of laboratory samples as the basis for modeled saltstone properties or the uncertainty in modeled saltstone properties. As described in the Technical Notes for MF 3.02, DOE has used analyses of laboratory samples to study saltstone variability.

NRC staff should compare any DOE measurements of field-emplaced samples with the corresponding measurements (e.g., hydraulic conductivity, diffusivity, density, porosity) of laboratory-produced samples. In addition, NRC staff should evaluate the results of laboratory tests in which the composition or curing conditions of saltstone samples are varied (e.g., studies similar to those described in the Technical Notes for MF 3.02). NRC staff should determine whether the range of compositions and conditions tested reflects the ranges of compositions and conditions expected in field-emplaced saltstone.

In addition to considering variations in saltstone composition and emplacement conditions, NRC staff should consider the effects of scale. For example, small laboratory-made or small-scale core samples may underestimate the effects of fracturing on the bulk properties of emplaced saltstone if emplaced saltstone has fractures that are not captured by small samples.

Technical Notes for MF 3.04: Effect of Curing Temperature on Saltstone Hydraulic Properties

Several DOE studies demonstrated the importance of cure temperatures on saltstone properties (i.e., SRNL-STI-2009-00184, SRNL-STI-2009-00810, SRNL-STI-2011-00665). In particular, samples cured at 60ºC (140ºF) had an average hydraulic conductivity of 8.0×10^{-7} cm/s, which is more than 500 times greater than similar samples prepared at 20ºC (72ºF) and 400 times greater than the base-case hydraulic conductivity assumed in the DOE 2009 PA. DOE suggested that those results may have overrepresented the effects of curing temperature because the samples cured at 60ºC (140ºF) also were cured in a low-humidity environment. However, DOE previously found that saltstone samples cured at 90ºC (190ºF) in closed containers cracked during curing, even though the samples did not dry (WSRC-TR-98-00337). Thus, curing at 90ºC (190ºF) does appear to impact saltstone cracking through at least one mechanism, in addition to drying.

During two onsite observation visits, DOE indicated that thermocouples are used within the disposal structure and saltstone to detect curing temperatures *(NRC, 2008a; NRC, 2011b)*. NRC staff will review the cure temperature profiles for emplaced saltstone when they become available. NRC staff should determine whether the hydraulic conductivity values DOE used in Case A or Case K of the DOE 2009 PA are expected to be good estimates of saltstone hydraulic conductivity, considering curing temperature profiles observed in the field.

A.4 Monitoring Area 4 – Waste Form Physical Degradation

While performing the activities related to Monitoring Area 4, NRC staff should closely coordinate the activities related to Monitoring Area 3. In particular, NRC staff should ensure that any information about initial saltstone cracking related to MF 3.02 and MF 3.04 are consistent with assumptions made in the projection of long-term saltstone fracturing.

Technical Notes for MF 4.01: Waste Form Matrix Degradation

The DOE Case A simulation did not consider physical degradation of saltstone. The Case A assumption that saltstone will not degrade physically during the performance period is based on a thermodynamic and mass balance analysis (WSRC-STI-2008-00236). In response to NRC questions *(NRC, 2008c)*, DOE characterized that research as an “initial step in trying to understand expansive phase precipitation during saltstone evolution” (SRR-CWDA-2009-00011, Rev. 0). That work was based on a simulation, implemented with the Geochemist’s

Workbench® geochemical code, of the formation of expansive mineral phases in saltstone due to exposure to rainwater or rainwater equilibrated with cement.

NRC staff should be familiar with the previous NRC technical review of the DOE mechanistic modeling work *(NRC, 2008c)* and NRC staff concerns about the use of that work as a basis for conclusions about long-term saltstone degradation (see NRC 2012 TER, Section 2.6.4.2). Those NRC staff concerns included: (1) lack of model validation with experimental results, (2) uncertainty in the initial mineralogy, (3) uncertainty in the amount of porosity that can be filled before degradation occurs, (4) neglect of metastable phases that could temporarily consume more porosity than projected with equilibrium calculations, and (5) neglect of the potential effects of organic additives or pozzolanic replacements. In response to those NRC staff concerns, DOE described ongoing and planned research to address saltstone matrix degradation (SRR-CWDA-2011-00044, Comment SP-1). NRC staff should review the development of that DOE research.

DOE Case A also neglected dissolution of salts and low solubility matrix phases. DOE responded to these concerns by indicating that degradation was modeled non-mechanistically in sensitivity cases. In Case K, DOE assumed that saltstone hydraulic conductivity increases from 1×10^{-8} cm/s to 1×10^{-6} cm/s as a logarithmic function of time (i.e., most of the increase occurs late in the performance period). In the Case K PORFLOW® model of radionuclide release, DOE assumed saltstone diffusivity increased from 1×10^{-7} cm^2/s to 5×10^{-6} cm^2/s during the performance period as a logarithmic function of time. (Although the DOE PORFLOW model of radionuclide release used an increasing diffusivity, as explained in NRC 2012 TER, Section 2.7.1.3, the DOE off-line calculation of saltstone oxidation used in the average K_d model of Tc release used a fixed diffusivity of 1×10^{-7} cm^2/s to model the movement of the oxidation front.) As a form of model support, DOE indicated that the assumed final hydraulic conductivity and diffusivity of saltstone were representative of site soils. However, in the NRC 2012 TER, NRC staff determined that none of the sensitivity analyses, including Case K, provided reasonable assurance that the planned salt waste disposal at the SDF met the performance objective of §61.41. Thus, NRC staff should consider the DOE bases for neglecting those additional potential sources of saltstone degradation. Alternately, NRC staff should consider whether any non-mechanistic assumptions of saltstone degradation that DOE uses as part of a new demonstration of compliance with §61.41 are likely to adequately represent saltstone degradation.

<u>Technical Notes for MF 4.02: Waste Form Macroscopic Fracturing</u>

NRC staff believes that the DOE Case A assumption that saltstone will be intact for 20,000 years is unrealistically optimistic, and that this DOE assumption is inconsistent with observations of existing cracks in SDS 4 saltstone (SRNL-ESB-2008-00017, SRR-CWDA-2011-00105). In the response to NRC RAI Comment SP-2 (SRR-CWDA-2011-00044), DOE indicated that it does not believe that there is evidence of these cracks extending below the surface of the monolith. However, in SRR-CWDA-2011-00105, DOE noted that the depth and thickness of those cracks is not known. Although it is unclear whether the cracks penetrate deeply enough to be hydraulically significant, cracks that do not penetrate saltstone still provide additional surface area for oxidation. Grout was first added to SDS 4 in 1996 and it has been added intermittently into SDS 4 since then.

Furthermore, in the DOE 2009 PA, DOE considered only sulfate attack and did not provide an adequate basis for neglecting other types of degradation, such as shrinkage cracking, steel corrosion-induced cracking, cracking due to settlement or earthquakes, and dissolution of salts and low solubility matrix phases. Of those mechanisms, NRC staff has DOE estimates of fracturing only for static and dynamic settlement of saltstone in SDS 4. NRC staff should evaluate DOE bases for neglecting those additional potential sources of saltstone degradation.

In a settlement analysis for SDS 4 (T-CLC-Z-00006), DOE assumed static settlement caused cracking in saltstone along SDS 4 construction joints at 9.1 m (30 ft) intervals and between the saltstone and disposal structure walls. Dynamic settlement was assumed to cause cracking at 15.2 m (50 ft) intervals. DOE concluded that those projected fractures may be pinched closed at one end and, therefore, may not represent water pathways. However, it is not necessary for a fracture to fully penetrate saltstone to increase the effective hydraulic conductivity because a network of fractures that do not fully penetrate saltstone, even if disconnected, may increase the rate of water flow through the waste form.

As discussed in the Technical Notes for MF 4.01, NRC staff is concerned about the DOE conclusion that saltstone degradation (i.e., either matrix degradation or fracturing) by expansive phases due to sulfate attack is unlikely. That DOE conclusion was based on geochemical modeling results that NRC staff determined were unsupported by comparisons with empirical data or observations.

In DOE Case K, DOE assumed that saltstone will form a full-width, through-going fracture every 10 cm within 10,000 years after closure. That degree of fracturing was presented as a non-mechanistic sensitivity analysis assumption. The DOE and NRC sensitivity analyses demonstrated that projected SDF performance is sensitive to the assumed rate and degree of saltstone fracturing (NRC 2012 TER, Section 2.13). NRC staff understands the difficulty in projecting long-term saltstone fracturing. It may be appropriate for NRC staff to consider a range of potential rates of saltstone fracturing, including different functional dependencies on time (e.g., logarithmic, quadratic, linear). Because there is little physical evidence available to support projections of saltstone fracturing for thousands of years after emplacement, it may be useful to use formal expert elicitation to develop these ranges (see NUREG-1563, “Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program”).

A.5 Monitoring Area 5 – Waste Form Chemical Performance

Technical Notes for MF 5.01: Radionuclide Release from Field-Emplaced Saltstone

Most of the experiments to study the leaching of radionuclides from saltstone conducted to date have reflected the bulk constituents of saltstone and simulated waste, but have not included admixtures used in the production of saltstone at the SPF. It is important to have measurements of leaching from field-emplaced saltstone in order to understand the effects of the admixtures as well as potential variability in the formulation and curing conditions in the field. DOE performed a study of leaching from a saltstone core sample obtained from SDS 4 (SRNL-STI-2010-00667). That study provided useful leaching information, although NRC staff had some concerns about the interpretations of some of the data obtained in this study. For example, in SRR-CWDA-02011-00044, Comment SP-19, DOE indicated that the new oxidizing saltstone Tc K_d of 10 mL/g used for Cases K and K2 was based on a single measurement of 12 mL/g for adsorption from a saltstone core from SDS 4 (SRNL-STI-2010-00667). The saltstone core sample was only partially oxidized, which means that blast furnace slag was likely still present in the solid. The release of Tc is affected not only by the redox character of the water used in the experiment, but also the redox character of the solid. The effect of any remaining blast furnace slag in the saltstone could be to continue to sequester a fraction of the

Tc, even if the water itself is oxidizing. Because the conceptual model for saltstone Tc release assumes that oxidizing conditions are in effect when all the reducing agent (i.e., blast furnace slag) in the grout is exhausted, experiments that use saltstone still containing blast furnace slag do not faithfully reflect the oxidizing conditions simulated in the PA.

Additionally, as discussed further under Technical Notes for MF 5.04, some of the leaching measurements in that study (e.g., for Sr) appear to represent solubility rather than sorption. The use of leaching experiments that are solubility limited to determine K_d values can lead to a significant underestimation in the modeled leaching. When evaluating the results of leaching experiments performed to address MF 5.01, it is important for NRC staff to confirm that the measured values do not reflect precipitation rather than sorption. The results of leaching experiments can be misleading and inappropriate to apply to transport modeling if the sorption is controlled by solubility in the experiment and can lead to a significant underestimation of the dose.

Technical Notes for MF 5.02: Chemical Reduction of Tc by Saltstone

The ability of saltstone to reduce and retain Tc is extremely important to the dose. The modeling of the reduction and sorption of Tc (i.e., K_d value for Tc) is one of the most risk-significant portions of the PA. That was identified as part of the Factors in the NRC 2005 TER and Revision 0 of the Monitoring Plan (i.e., Factor 1-Oxidation of Saltstone and Factor 3-Model Support) and was also tracked as Open Issue 2009-1. The K_d values assumed for Tc under reducing conditions in certain DOE analyses and experimental measurements of those values are summarized in Table A-2.

As is discussed in more detail in NRC 2012 TER, Section 2.7.4.2, NRC staff determined that the K_d values assigned in the PA analyses for Tc under reducing conditions were not adequately supported. The Tc K_d measurements for simulated saltstone that have been performed to date have either not supported a Tc K_d value as high as what was assumed or for saltstone under reducing conditions (i.e., 500 mL/g or 1000 mL/g), or the experiments have been performed in an atmosphere that is significantly different than for the actual as-emplaced saltstone.

Furthermore, the use of a reducing K_d value requires that oxygen is essentially absent from the system, because studies have shown that even very small quantities of oxygen (i.e., 30 ppm O_2 to 60 ppm O_2), in a system oxidize Tc in saltstone (SRNL-STI-2010-00668). DOE has not demonstrated that oxygen will be absent from the water-saltstone system when release is possible. Although NRC staff determined that there is also insufficient basis for the Tc K_d value of 500 mL/g assumed in Cases K1 and K2, that K_d value is more defensible than the value of 1,000 mL/g. Because NRC staff does not find that adequate support exists for the Tc K_d value under reducing conditions, NRC staff should assess the development of information regarding the chemical reduction of Tc(VII) to Tc(IV) in saltstone under the range of conditions to which saltstone is expected to be subjected during the performance period, the K_d value used to represent the Tc, and the maintenance of this reduction over time.

Table A-2: Saltstone K_d Value for Tc under Reducing Conditions

	K_d Value (mL/g)	Notes
Modeled Values		
Case A	1000	
Case K	1000	
Case K1	500	
Case K2	500	
Measured K_d Values for Simulated Saltstone		
SRNS-STI-2008-00045	6.5 to 91.3	Measurements made in solutions of $Ca(OH)_2$ and $CaCO_3$ that were purged with N_2. DOE noted that the quality of the spectra in these measurements was compromised by a U-233 shift.
SRNL-STI-2009-00636	32 [2] (9.1 to 56)	Experiments performed in a glovebox with a 2% H_2 environment. Measurement taken after 4 d and may not have reached steady state.[1]
SRNL-STI-2010-00668	711 [2]	Experiments performed in a glovebox with a 2% H_2 environment. Measurement taken after 22 d.[1]
	518 [2]	Experiments performed in a glovebox with a 2% H_2 environment. Measurement taken after 56 d. [1]
SRNL-STI-2011-00716	1258 [2] (757 to 1759)	Experiments performed in 0.1% H_2 atmosphere in the presence of a palladium catalyst to convert O_2 to water through reaction with H_2). Measurement taken after 56 days.[1]
Measured K_d Values for Saltstone Core Sample		
SRNL-STI-2010-00667	139	Leached under N_2 using a $Ca(OH)_2$ solution. DOE noted that the nitrogen glovebag contained small amounts of O_2. Measurement taken after 20 d.

(1) Measurements performed using a solution of $CaCO_3$ unless otherwise reported

(2) Measurements performed in an environment containing H_2 (g)

Technical Notes for MF 5.03: Reducing Capacity of Saltstone

The DOE Case K model and NRC sensitivity analyses demonstrated the importance of saltstone reducing capacity to the projected Tc release rate. In the NRC 2012 TER, Section 2.6.4.3, NRC staff evaluated the modeling of chemical properties of the waste form, including the initial reducing capacity of saltstone and the expected evolution of redox conditions over time. NRC staff determined that there was not adequate support for the reducing capacity for saltstone or for the projected evolution of redox conditions.

NRC staff is concerned that these DOE measurements of saltstone reducing capacity (SRNS-STI-2008-00045) indicate the reducing capacity of saltstone is equivalent to the reducing capacity measured for pure blast furnace slag, even though the saltstone sample only contained 23 weight percent blast furnace slag. NRC staff also is concerned that, in actual field conditions, only a fraction of the slag will be accessible for reaction with the infiltrate and that the reactive surface area and reducing capacity of saltstone emplaced in the field are likely to be much smaller than that of finely ground laboratory samples. Also, the reducing capacity of saltstone used in the model is based on laboratory measurements using slag samples that were finely ground to increase the reactive surface area (SRNS-STI-2008-00045). In the Case K, K1, and K2 analyses, DOE assumed that the saltstone reducing capacity was reduced by a factor of four to account for the saltstone composition (i.e., ~25% blast furnace slag). NRC staff determined that the resulting decreased E_h transition time in Case K more reasonably reflects the expected evolution of the saltstone chemical environment, but it is still necessary to obtain better information on the reducing capacity of saltstone due to its significant effect on dose.

There is also uncertainty in the chemical modeling used to estimate the E_h transition times. Those model results have not been validated by comparison to experimental or other data independent of model calculations. Additionally, there is uncertainty in the initial mineralogy assumed in the models. The geochemical modeling of E_h transitions also assumed all the reducing capacity in the saltstone is available for reaction with the infiltrate. The reactive surface area and reducing capacity of saltstone emplaced in the field are likely to be much smaller than that of finely ground laboratory samples. Thus, the longevity of reducing chemical conditions and the timing of release of redox-sensitive elements such as Tc, Pu, and Np, may be overestimated.

Technical Notes for MF 5.04: Certain Risk-Significant K_d Values for Saltstone

In Section 2.7.4.2 of the NRC 2012 TER, NRC staff evaluated the potentially risk-significant K_d values assumed for saltstone and cementitious materials in the DOE analyses. NRC staff determined that there was insufficient support for the K_d values assumed for Ra, Se, and Sr.

NRC staff should consider whether the formulation used in the experiments is consistent with saltstone. Additionally, NRC staff should consider whether the experiments represent leaching of the radionuclide from saltstone or sorption of the radionuclide onto saltstone. The measured retention of the radionuclide may be higher in experiments where the radionuclide is present in the saltstone while it is setting (as opposed to experiments where the radionuclide is sorbed onto simulated saltstone that has already set) because it may become more highly incorporated into the waste form. Also, as discussed in the Technical Notes for MF 5.01, the amount of leaching may be underestimated if the leaching experiments are solubility limited.

In DOE Case A and many of the sensitivity cases, the dose from Ra-226 contributed significantly to the dose. The projected dose from Ra-226 in DOE Case K was significantly less due to the use of a revised inventory for Ra-226 and its ancestors. Case K also used revised K_d values for Ra in saltstone and other cementitious materials. The K_d values assumed for Ra are summarized in the NRC 2012 TER, Table 2.7-5. In Case A and the other PA cases, the Ra cementitious material K_d values for oxidizing conditions were based on the literature and the reducing values in the PA were set equal to Sr values. The contrasting sources of information were responsible for the unexpectedly large differences in the two sets of values for an element that is generally insensitive to oxidation state. In Case K, the K_d values for reducing conditions were set to the same values as for oxidizing conditions because Ra sorption is not expected to be redox sensitive. Because the Ra K_d is potentially significant to dose and directly applicable data on the Ra sorption in saltstone are lacking, obtaining measurements of sorption of Ra in saltstone is important. As discussed in Section 3.5.4 of this document, if adequate support is developed for the revised inventory for Ra and its ancestors, then the Ra-226 dose may no longer be significant and this aspect of MF 5.04 will not need to be addressed. Thus, NRC staff should consider the information provided for MFs 1.01 and 1.02 when evaluating MF 5.04.

The K_d values assumed for Se in saltstone and other cementitious materials are described in NRC 2012 TER, Table 2.7-6. As discussed in more detail NRC 2012 TER, Section 2.7.4.2, NRC staff determined that the K_d values assumed for Se for cementitious materials in reducing conditions appear to be reasonable, but there was not an adequate basis for the K_d values assumed for oxidizing conditions. In particular, NRC staff determined that the DOE response (SRR-CWDA-2011-00044, Comment C-4) to a follow-up to an original RAI response

(SRR-CWDA-2010-00033, Comment C-4) did not provide sufficient information to show that selenite would be the important species under oxidizing conditions or that selenate, the expected oxidized species, would have K_d values in excess of the 30 to 79 mL/g range measured by DOE for simulated saltstone and disposal structure concrete (SRNS-STI-02008-00045). In DOE Cases K, K1, and K2, DOE used an oxidized old age Se K_d of 30 mL/g (SRR-CWDA-2011-00044, Table PA-8.7), and Se-79 did not become an important dose contributor. However, in those cases, the saltstone did not transition to old age within 20,000 years and DOE did not test the sensitivity of Se dose to a lower Se K_d (i.e., lower than the value of 300 mg/L assumed in the DOE analyses) for oxidized middle age conditions. Therefore, NRC staff determined that additional experimental measurements of the Se K_d for saltstone under oxidizing conditions were important.

The K_d values assumed for Sr in saltstone and other cementitious materials are described in the NRC 2012 TER, Table 2.7-6. In the TER, NRC staff determined that the K_d values assumed in the PA cases for Sr in saltstone were acceptable. In DOE Cases K, K1, and K2, the K_d values for Sr for saltstone were changed upward on the order of two to three orders of magnitude to 1,000 mL/g. Those revised values have the potential to significantly limit the modeled Sr release, which could significantly affect the projected dose to an inadvertent intruder. The new values were based on desorption experiments on an actual ground saltstone core sample from SDS 4, which had evidence for partial oxidation (SRNL-STI-2010-00667). That core study is discussed in more detail under Technical Notes for MF 5.01. It is not clear to NRC staff that sufficient data have been obtained to support using those new derived Sr values in PAs. For example, the investigators who performed that research speculated that the dissolved Sr measurements may, in fact, be controlled by $SrSO_4$ solubility rather than sorption. There is a suggestion in the document that $SrSO_3$ could also be a controlling solid, perhaps under more reducing conditions. If the amount of desorption observed in a leaching experiment is controlled by solubility, then the experimental artifact will result in the calculated K_d value being artificially high. Thus, DOE needs to provide more information on Sr leaching from saltstone to support the revised Sr K_d values.

Technical Notes for MF 5.05: Potential for Short Term Rinse-Release from Saltstone

Experiments of leaching of radionuclides from saltstone have shown a high initial “rinse-release” of radionuclides. As discussed in Section 3.5.5 of this document, it is not clear to NRC staff if that initial release is due to an experimental artifact or if it is due to radionuclides that are not fully encapsulated in the saltstone waste form (e.g., radionuclides that are present in the pore solution).

In a report by Pablan *et al.* (2012), the initial leaching of Tc from simulated saltstone was high. Similarly, in a report by Tallent *et al.* (1987), the initial leaching of Tc from a cement-based waste form was higher than at later times. Tallent *et al.* concluded that the initial leaching was considered to be a surface-related or wash-off problem and was not indicative of the long-term leach properties of the technetium in the grout specimens.

Because the dose during the first few pore volumes that flow through the waste form could be high if the radionuclides are not fully encapsulated in the saltstone, it is important to understand if the “rinse-release” observed in these experiments is an experimental artifact or if it represents the expected leaching from the waste form.

A.6 Monitoring Area 6 – Disposal Structure Performance

Technical Notes for MF 6.01: Certain Risk-Significant K_d Values in Disposal Structure Concrete

In DOE Cases K, K1, and K2, DOE used much higher K_d values for Ra on reducing cementitious materials than in Case A (SRR-CWDA-2011-00044, Table PA-8.6). The values were increased to match the values used in oxidizing cementitious materials because Ra sorption was not expected to be redox sensitive. NRC staff agrees Ra sorption is not expected to be redox sensitive. The values used for oxidizing and reducing conditions were based on literature data (DOE 2009 PA, Table 4.2-18). If Ra-226 continues to be a risk-significant radionuclide, then Ra-226 sorption should be based on experiments of Ra sorption to samples of disposal structure concrete. DOE indicated that it would consider future studies on the transport properties of Ra in disposal structure concrete (SRR-CWDA-2010-00033 and SRR-CWDA-2011-00044, Comment SP-14). However, in response to the NRC second RAI, DOE significantly reduced the projected inventories of Ra-226 and its ancestors, U-234 and Th-230. The importance of MF 6.01 depends on the DOE support for this inventory change. Due to that interdependence, NRC staff should consider MF 6.01 in conjunction with MF 1.01.

As discussed in the NRC 2012 TER, Section 2.7.4.2, and in the Technical Notes for MFs 5.04 and 7.01, the risk-significance of Se-79 disposal is unclear to NRC staff. Se-79 was not identified as causing a significant portion of the dose in the DOE Case A or alternate deterministic sensitivity cases. DOE provided the results of a sensitivity analysis performed using the probabilistic GoldSim® model in which the Se K_d values for the sandy and clayey soils were set to 0 mL/g (SRR-CWDA-2010-00033). The results of that sensitivity analysis showed that changing the Se K_d values resulted in a small absolute increase in dose, but a large relative increase in the dose derived from Se-79. Based on that assessment, combined with the uncertainty in other key parameters related to Se release and transport (e.g., MF 5.04, MF 7.01), NRC staff determined that the disposal structure K_d values for Se may be important to dose.

DOE also does not appear to have an adequate technical basis for the K_d values used to model Se sorption in oxidized middle age (300 mL/g) or oxidized old age (150 mL/g) disposal structure concrete. It is unclear to NRC staff whether changes to those values will change the projected risk significance of Se-79. Although Se-79 was not an important dose contributor in Case K, the disposal structure concrete was not assumed to transition to old age until 7,300 years after closure. DOE did not test the sensitivity of the Se dose to a lower Se K_d for oxidized middle age conditions.

NRC staff should evaluate the development of information to support the DOE K_d values for Se. NRC staff should pay particular attention to the oxidation state of Se in the experiments used as the basis for K_d values. NRC staff expects that disposal structure concrete K_d values for selenate (i.e., the expected oxidized species) should be lower than the values that DOE used in the PA. Specifically, DOE measured K_d values of 30 to 79 mL/g for selenate on samples of simulated saltstone or disposal structure concrete (SRNS-STI-2008-00045).

Technical Notes for MF 6.02: Technetium Sorption in Disposal Structure Concrete

As discussed in the NRC 2012 TER, Section 2.13, NRC analyses of intermediate results from the DOE Case K, K1, and K2 PORFLOW™ model demonstrated the importance of Tc retention in disposal structure concrete to the timing and magnitude of the dose from fractured saltstone. The modeled re-concentration of Tc in disposal structure concrete occurs because the disposal structure floors and walls were modeled with much higher K_d values than saltstone as saltstone becomes more oxidized (e.g., at 10,000 years after closure, saltstone is modeled with a K_d of 0.8 mL/g, as compared to the floor of disposal structures similar to SDS 2A with a K_d of 406 mL/g, as compared to the walls of disposal structures similar to SDS 2A with a K_d of 388 mL/g, as compared to the SDS 4 floor with a K_d of 388 mL/g, and as compared to the walls of SDS 4 with a K_d of 228 mL/g). NRC staff does not believe that modeled behavior is realistic because: (1) the disposal structure concrete is expected to have about 40% of the specific reducing capacity of saltstone due to its lower blast furnace slag concentration; (2) the disposal structure floors and walls are more exposed to environmental conditions than saltstone is and NRC staff expects them to fracture within 10,000 years; (3) water flowing through fractures in the disposal structure floors and walls would be expected to create oxidized conduits where Tc would not be expected to be re-reduced and immobilized after release from the saltstone, and (4) water may flow out of disposal structures through joints and consequently may not interact significantly with the disposal structure concrete.

The two primary modeled effects of this re-concentration in the disposal structure floor and walls in DOE Case K were to delay the projected peak dose by approximately 5,000 years and to significantly reduce the magnitude of the projected peak dose. The projected peak dose was reduced because it is very sensitive to the rate at which Tc is released into the environment and the modeled Tc release from the disposal structures is much slower than the modeled Tc release from the saltstone.

Table A-3: Saltstone K_d Value for Tc under Oxidizing Conditions

	K_d Value (mL/g)	**Notes**
Modeled Values		
Case A	0.8	
Case K	10	
Case K1	0.8	
Case K2	10	
Measured K_d Values for Simulated Saltstone		
SRNS-STI-2008-00045	0.16 to 0.93	Measurements made in solutions of $Ca(OH)_2$ and $CaCO_3$.
	-0.02 to 0.25	Simulated saltstone was partially oxidized prior to experiment. Measurements made in solutions of $Ca(OH)_2$ and $CaCO_3$.
SRNL-STI-2009-00636	5.0 (3.1 to 6.3)	Measurement taken after 4 d and may not have reached steady state. [1]
Measured K_d Values for Saltstone Core Sample		
SRNL-STI-2010-00667	12	Leached using a $Ca(OH)_2$ solution. Purged with air continuously. Measurement taken after 20 d.

[1] Measurements performed using a solution of $CaCO_3$, unless otherwise reported.

NRC staff should evaluate DOE model support for Tc sorption in oxidized and reduced disposal structure concrete. Tc sorption in disposal structure concrete may differ from Tc sorption in saltstone because of differences in the formulation of the two cementitious materials, including differences in the reducing capacities. In addition, K_d values based on leaching Tc from saltstone (i.e., when Tc is initially incorporated into the saltstone) may not be applicable to estimates of Tc adsorption to disposal structure concrete. NRC staff should evaluate when different physical processes govern Tc dissolved concentrations (e.g., dissolution of initially precipitated Tc as compared to adsorption of initially dissolved Tc). NRC staff should be familiar with previous DOE studies of Tc sorption on reduced and oxidized saltstone and the concerns DOE and NRC staff have about each study (see both Table A-2 and Table A-3) because DOE has used experiments with saltstone as a basis for its estimates of Tc sorption to disposal structure concrete. NRC staff also should be familiar with those saltstone sorption studies because similar experimental concerns may be encountered in studies of Tc sorption onto disposal structure concrete.

NRC staff should consider the potential effects of fractures in the disposal structure concrete or joints between disposal structure components (e.g., floors and walls) on Tc sorption. In particular, Tc moving through fractures could encounter faster water flow, fewer sorption sites, and (on average) more oxidizing conditions than Tc moving through intact disposal structure concrete.

<u>Technical Notes for MF 6.03: Performance of Disposal Structure Roofs and HDPE/GCL Layers</u>

As discussed in the NRC 2012 TER, Section 2.4.3.1, the lower lateral drainage layer diverts a significant amount of water from the disposal structures; thereby, reducing radionuclide release into the groundwater. For example, in the DOE Case A, approximately 99.9% of the water is modeled as being shed around the disposal structures at 10,000 years after SDF closure (SRR-CWDA-2011-00044, Comment IEC-8).

NRC staff expects that support for the performance of the lower lateral drainage layer will include support for the long-term hydraulic conductivities of the disposal structure roofs, HDPE/GCL layers overlying the disposal structures similar to SDS 2A, and the sand layer of the lower lateral drainage layer. If DOE determines that the values used in the DOE 2009 PA are not supportable, then NRC staff should request that DOE recalculate the expected amount of water diversion based on the supportable hydraulic properties.

NRC staff should be aware of a DOE sensitivity analysis to assess the risk significance of filter fabric by doubling the modeled infilling of the drainage layers (SRR-CWDA-2011-00044, Comment PA-10). The DOE analysis demonstrated that modeled shedding of the water around the disposal structures was much more sensitive to assumptions about the disposal structure roofs (for SDS 1 and SDS 4) and the HDPE/GCL composite layer above the disposal structures similar to SDS 2A than it was to assumptions about drainage layer infilling. For SDS 4 and for a disposal structure similar to SDS 2A, doubling the infilling of the drainage layers increased the Darcy velocity through saltstone by approximately a factor of three or less at 10,000 years in Case A and caused smaller increases in Case K. In comparison, the modeled Darcy velocity through saltstone at 10,000 years after closure is between a factor of 100 and 1,000 times greater in Case K than it is in Case A.

In the NRC 2012 TER, NRC staff agreed with the DOE assessment that the degradation of disposal structure concrete and saltstone is more significant to risk than the degradation of the drainage layers. However, a factor of two to three in difference in the Darcy velocity is more significant if projected doses approach the relevant dose limit. NRC staff should be aware that aspects of MF 6.03 are closely related to aspects of MF 2.01, 6.04, MF 9.01, and MF 9.02.

<u>Technical Notes for MF 6.04: Disposal Structure Concrete Fracturing</u>

NRC analyses of intermediate results from the DOE Case K PORFLOW™ model demonstrated the importance of Tc retention in disposal structure concrete to the timing and magnitude of the projected dose from fractured saltstone. NRC staff expects fractures in the disposal structure to be important to SDF performance for the following three reasons: (1) fractures could increase water flow through the disposal structures; (2) there may be reduced sorption in fractures, as compared to intact concrete; and (3) there may be increased oxidation of the disposal structure concrete due to dissolved oxygen in infiltrating water or oxygen in soil gas.

In Case A, DOE assumed fracturing of the SDS1 and SDS 4 walls with minimal degradation of SDS 1 floor, SDS 4 floor, or any of the concrete in disposal structures similar to SDS 2A. In the DOE 2009 PA cases (i.e., cases other than K, K1, and K2), the degradation of the disposal structure concrete was based on sulfate attack modeling performed with the STADIUM® code. As discussed in the NRC 2012 TER, NRC staff was concerned about additional potential degradation mechanisms, including alkali-silica reaction; carbonation; chloride-induced corrosion; calcium leaching; microbial degradation; freeze-thaw cycles; and cracking from seismic events, settlement, and external static loading. NRC staff should be aware of the DOE discussion of those degradation mechanisms in the second RAI response (SRR-CWDA-2011-00044, Comment VP-2). In the NRC 2012 TER, Section 2.5.3.2, NRC staff determined that additional consideration should be given to the potential coupling of multiple degradation mechanisms. In particular, NRC staff determined that DOE did not account for chemical degradation mechanisms that tend to decrease the strength of cementitious materials (e.g., sulfate attack, rebar corrosion, alkali-silica reaction) in the DOE structural integrity analysis (T-CLC-Z-00006).

In the NRC 2012 TER, Chapter 5, NRC staff had concerns about some aspects of the DOE settlement and structural integrity analysis. Specifically, geotechnical investigations for SDS 2A, SDS 2B, SDS 3A, SDS 3B, SDS 5A, and SDS 5B projected settlement values greater than the American Concrete Institute (ACI) standard (ACI-372) (K-ESR-Z-00001, K-ESR-Z-00002). Because neither the standard nor the DOE analysis addressed the consequences of exceeding those criteria, the risk-significance of this finding is unclear to NRC staff. Accordingly, NRC staff should follow the DOE development of information related to settlement of the SDF due to static loading.

As described in the NRC 2012 TER, Section 2.5.2.1, the fraction of disposal structure concrete oxidized as a function of time could be calculated from the K_d values that DOE assigned to the disposal structure concrete. A significant fraction of the disposal structure concrete was modeled as remaining reduced during a 20,000 year analysis time (e.g., less than 30% oxidation of the concrete in disposal structures similar to SDS 2A within 10,000 years).

NRC staff is not currently aware of any engineered or natural analogues to validate projections of disposal structure fracturing for several thousand years after site closure. NRC staff should be aware of the development of models of concrete degradation and any existing model support for those models (e.g., experimental validation of short-term projections and support for longer-term results by accelerated aging of materials). In particular, NRC staff should be familiar with work performed by the Cementitious Barriers Partnership, including work used to support the DOE 2009 PA and new Cementitious Barriers Partnership work in that area. NRC staff should use that information to evaluate the DOE estimates of disposal structure fracturing during the performance period.

Technical Notes for MF 6.05: Integrity of Non-cementitious Materials

Long-term properties of non-cementitious materials in disposal structures are of interest to NRC staff primarily because of the potential formation of fast pathways through the disposal structures due to degradation of the non-cementitious materials. Like fractures in disposal structure concrete (see Technical Notes for MF 6.04), fast pathways through disposal structures due to degradation of non-cementitious materials are a concern to NRC staff because they have the potential to lead to increased water flow, decreased radionuclide contact with sorption sites, and increased oxidation. NRC staff should evaluate DOE development of support for the long-term durability of non-cementitious materials used in the disposal structures under the relevant service conditions (e.g., in a humid, alkaline, high sulfate environment).

A.7 Monitoring Area 7 – Subsurface Transport

Technical Notes for MF 7.01: Certain Risk-Significant K_d Values in Site Sand and Clay

NRC staff should evaluate the development of subsurface K_d values for Se. In the NRC TER, Section 2.7.4.3, NRC staff evaluated the subsurface K_d values for elements that had the potential to be risk-significant (i.e., Tc, I, Sr, Ra, Se). NRC staff determined that the subsurface K_d values assumed by DOE for those elements were well supported, except for Se.

NRC staff determined that the K_d value of 1000 mL/g assumed in the PA for Se in sand and clay was not adequately supported because that value was representative of Se sorption in a low pH soil. An analysis described by DOE in the RAI response (SRR-CWDA-2011-00044), estimated that the Se K_d values will decrease sharply (i.e., there is less sorption) as the pH increases above pH 6; and will decrease an order of magnitude as the pH approaches 7. In the RAI response, DOE also provided pH readings for site groundwater samples. Of those samples, 42% (30 of 72) had a pH value over 6 and 8.3% (6 of 72) had a pH over 7. Furthermore, the pH of water near the disposal structures may be elevated by the presence of saltstone and the cementitious disposal structures. Additionally, NRC staff believes that the multiple measurements of a Se K_d of precisely 1,041 mL/g in the original report cited by DOE (WSRC-STI-2006-00037, Table 6) appears unusual and is suggestive of an experimental artifact that has not been explained by DOE.

A.8 Monitoring Area 8 – Environmental Monitoring

Technical Notes for MF 8.01: Leak Detection

There are no Technical Notes for MF 8.01.

Technical Notes for MF 8.02: Groundwater Monitoring

In the area of the SDF, the hydrological system consists of three aquifers of interest: (1) the water table or UTR aquifer, which is split into upper and lower zones; (2) the Gordon aquifer; and (3) the Crouch Branch aquifer. The UTR and Gordon aquifers are expected to be impacted by radionuclides from the SDF. Contamination is not expected to affect the deeper Crouch Branch aquifer; however, because of an upward flow gradient between the Crouch Branch and Gordon aquifers near Upper Three Runs Creek. The northwest half of the SDF straddles a groundwater divide between Upper Three Runs Creek and McQueen Branch causing SDF contaminants to discharge to either creek depending on the SDF source location (DOE 2009 PA, page 172). Additional useful background information on the SDF hydrological system and far-field modeling approach and results was presented in several PA sections (i.e., Sections 3.1.5, 4.2.3.1.3, 4.2.3.2.5, 4.3.1.2). NRC staff evaluation of the far-field portion of the PA in Section 2.8 of the NRC 2012 TER also contains useful information regarding the groundwater at the site.

According to the Groundwater Monitoring Plan for the Z-Area Saltstone Disposal Facility (WSRC-TR-2005-00257), DOE plans to monitor the groundwater in three water table wells located downgradient of SDS 4 (designated ZBG3, ZBG4, and ZBG5) as well as in one background well (designated ZBG1). In addition, three wells (designated ZBG6, ZBG7, and ZBG8) have been installed downgradient of SDS 1. In the future, if new disposal structures are constructed downgradient of the current wells, then new wells will be installed within 30 m [100 ft] downgradient of the new disposal structures. See below Figure A-1 containing a more recent image of monitoring well locations at the SDF, which came from “Saltstone Disposal Facility Class 3 Landfill Groundwater Monitoring Midyear Report for 2012” (SRNS-TR-2012-00125), dated July 2012.

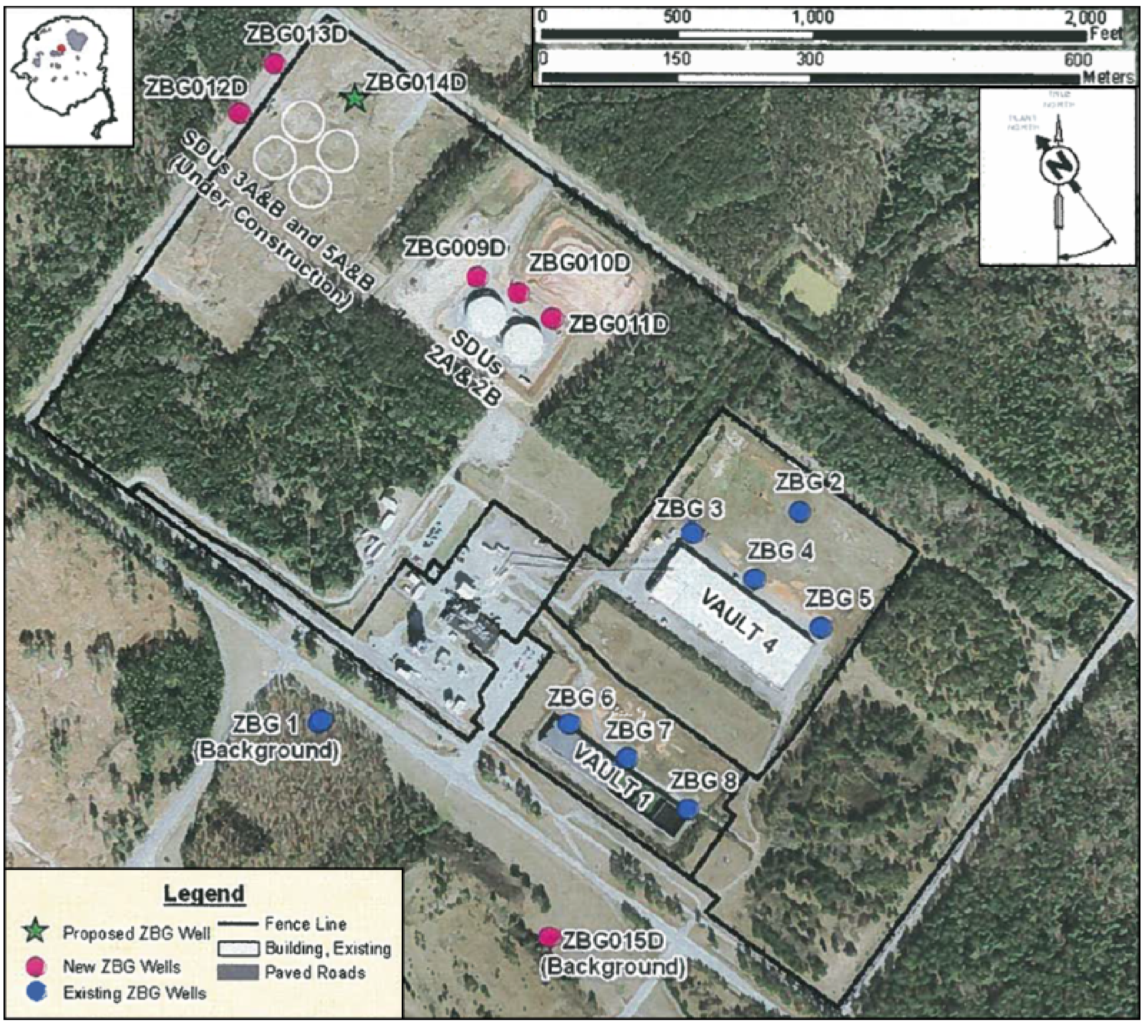

Figure A-1: Location of Z-Area Wells (from SRNS-TR-2012-00125)

NRC staff reviewed aspects of the groundwater monitoring program during onsite observation visits in October 2007 and March 2008 *(NRC, 2008a; NRC, 2008b)*. During the October 2007 visit, NRC staff reviewed procedures relating to groundwater monitoring well installation, sampling methods, and sample packaging. NRC staff also reviewed groundwater sample results. During the March 2008 onsite observation visit, NRC staff reviewed the 2007 groundwater monitoring program results for eight groundwater monitoring wells installed in or near the SDF (WSRC-TR-2008-00001). During the visit, NRC staff noted that the groundwater monitoring report indicated the existence of a groundwater divide on the Z-area. NRC staff also noted elevated tritium observed in wells located downgradient of SDS 1. DOE contractor staff indicated that the source of the tritium was inconclusive at that time. NRC staff also reviewed the nitrate concentrations and determined that the observed nitrate concentration in downgradient wells was similar to measurements from the background well, ZBG-1.

NRC staff should continue to review the groundwater monitoring program as new disposal structures are constructed and new wells are installed. NRC staff should consider whether the wells are adequate to assess whether leaching from saltstone has occurred. NRC staff should consider if the location of the background well is appropriate to assess leaching from saltstone and the resulting plumes. NRC staff should evaluate the significance of any impacts of the groundwater divide on whether the well locations are adequate. In reviewing the well locations, NRC staff should consider water elevation data. NRC staff should also request groundwater monitoring data on an annual basis and review it for increases in the concentration of radionuclides or saltstone indicators.

A.9 Monitoring Area 9 – Site Stability

Technical Notes for MF 9.01: Settlement Due to Increased Overburden

As discussed in the NRC 2012 TER, Chapter 5, the magnitude of both uniform and differential settlement assumed in the DOE 2009 PA was significantly less than the settlement projected in recent DOE geotechnical investigations (K-ESR-Z-00001, K-ESR-Z-00002). In the NRC 2012 TER, Section 5.2, NRC staff discussed that DOE had not demonstrated whether settlement due to static loading may disrupt the performance of synthetic materials and drainage layers. In addition, relevant American Concrete Institute (ACI) standard (ACI-372) for the disposal structures suggests that the maximum recommended limits for uniform and differential settlement are 15 cm (6 in) and 5.3 cm (2.1 in), respectively, which are exceeded by the projected settlement for SDS 2A, SDS 2B, SDS 3A, SDS 3B, SDS 5A, and SDS 5B. The risk-significance of that is unclear because neither the ACI standard nor the DOE analysis addresses the consequences of exceeding those criteria. NRC staff should evaluate: (i) geotechnical studies, (ii) settlement data collected during SDF closure operations, and (iii) settlement data collected from analogous sites.

Technical Notes for MF 9.02: Settlement Due to the Dissolution of Calcareous Sediment

In the NRC 2012 TER, Section 5.2, NRC staff discussed the DOE evaluation of the effects of consolidation of the soft zones at the SDF. However, the DOE analyses of the potential settlement due to consolidation of these zones did not account for the potential removal of subsurface material (e.g. dissolution of subsurface minerals), which resulted in subsidence observed at SRS. There is evidence of sinks near the SDF that are much more significant than the DOE analysis of consolidation of the current soft zones suggests. NRC staff should evaluate and focus on: (1) processes that have resulted in the formation of sinks at SRS; (2) the potential for these processes to affect site stability throughout the performance period; and (3) the potential dose consequences from subsidence related to dissolution of calcareous sediment. The potential for consolidation of soft zones to affect site stability is most closely related to §61.44, whereas potential dose consequences are most closely related to §61.41 and §61.42.

A.10 Monitoring Area 10 – Performance Assessment Model Revisions

There are no Technical Notes for MFs 10.01 through 10.13.

A.11 Monitoring Area 11 – Radiation Protection Program

Technical Notes for MF 11.01: Dose to Individuals During Operations

When monitoring the dose to individuals during operations, NRC staff should review the DOE dose reports annually to assess whether the doses are consistent with the limits in 10 CFR Part 20 and are ALARA. NRC staff should focus the review primarily on the dose to the radiation workers because it is expected to be much larger (relative to the applicable dose limit) than the dose to members of the public due to the distance from the SDF to the site boundary. There may be some members of the public who come onsite, such as SRS visitors (including visitors to the SDF), hunters, and non-radiation workers, but the dose to these individuals is still expected to be relatively small compared to the radiation workers due to the limited time spent by these individuals near the radioactive materials.

In October 2007 *(NRC, 2008a)*, NRC staff interviewed the DOE contractor environmental monitoring personnel, reviewed records and radiological control documents associated with SDF operations, and reviewed associated worker dose records to verify that the DOE radiation protection program was in place for operations at the SDF. Based on this review, NRC staff determined that DOE has an adequate radiation protection program in place for SPF and SDF operations. NRC staff should periodically confirm that the DOE radiation protection program remains adequate, particularly if major changes are made to the salt waste disposal process or the radiation protection program. Additionally, as part of monitoring the potential dose to a member of the public during operations, NRC staff should confirm that the site access control is sufficient.

Additional information that may be useful to NRC staff in reviewing the dose to individuals during operations can be found in NUREG-1736 (Consolidated Guidance: 10 CFR Part 20 – Standards for Protection Against Radiation) and in several NRC Regulatory Guides (RGs), such as RG 8.2 (Administrative Practices in Radiation Surveys and Monitoring), RG 8.34 (Monitoring Criteria and Methods to Calculate Occupational Radiation Doses), and RG 8.40 (Methods for Measuring Effective Dose Equivalent from External Exposure). (Note that this is not an exhaustive list of potentially relevant guidance.) Additional information that may be useful in assessing whether the ALARA requirement is met can be found in RG 8.10 (Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable).

Technical Notes for MF 11.02: Air Monitoring

NRC staff should review the air monitoring data associated with the disposal of salt waste to determine whether the airborne emissions are consistent with the requirements of 10 CFR Part 20. The ALARA requirement of 20.1101(b) and 20.1101(d) specify a constraint of 0.10 mSv/yr (10 mrem/yr) from airborne emissions, excluding Rn-222 and its daughters, to a member of the public likely to receive the highest dose. NRC RG 4.20 (Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors) includes when determining the member of the public likely to receive the highest dose from airborne emissions of radioactive material to the environment, nonresidents within the facility boundary do not need to be considered. However, the total dose from airborne emissions and other sources to a member of the public within the facility boundary still must be less than 1 mSv/yr (100 mrem/yr) and ALARA. Similarly, the total dose from airborne emissions and other sources to a radiation worker must be consistent with the 50 mSv/yr (5 rem/yr) radiation worker dose limit.

In October 2007 *(NRC,2008a)* and March 2008 *(NRC, 2008b)*, NRC staff conducted onsite observation visits of the air monitoring program for the SPF stack and SDS 4. During those visits, NRC staff found that the air effluent sampling results for SDS 4 during filling operations indicated that doses to nearby workers and members of the public from air effluents were well below applicable limits. If any major changes are made to the salt waste disposal process that could result in an increased release of airborne emissions, then NRC staff should reevaluate the potential dose from airborne emissions to a radiation worker or a member of the public located onsite to assess whether releases from the disposal structures and SPF remain acceptable. Similarly, if any major changes are made to the air monitoring program, then NRC staff should reevaluate the program to ensure that the program continues to adequately monitor the air effluents.

When assessing whether the dose to an off-site member of the public from the air pathways is less than 0.10 mSv/yr (10 mrem/yr), NRC staff should focus on sample results from air sampling stations located at the site boundary. The site boundary air monitoring stations may have inputs from activities at SRS other than salt waste disposal. Therefore, if the monitoring data indicate that the dose to a member of the public could be greater than 0.1 mSv/yr (10 mrem/yr), an evaluation should be performed to assess how much of the dose is from salt waste disposal operations. That assessment could include the review of data from air monitoring stations located on site and stack effluent monitoring.

Additional information that may be useful to NRC staff in assessing the air monitoring program at SRS can be found in RG 4.20 (Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors.