SRR-CWDA-2019-00128
Revision 0

FY2019 ANNUAL REVIEW
SALTSTONE DISPOSAL FACILITY (Z AREA)
PERFORMANCE ASSESSMENT

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EXECUTIVE SUMMARY

The Saltstone Disposal Facility (SDF) presently consists of Saltstone Disposal Units (SDUs) 1, 2, 3, 4, 5, and 6 as described in the *Performance Assessment for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2009-00017), the *FY2013 Special Analysis for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2013-00062), the *FY2014 Special Analysis for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2014-00006), and the *FY2016 Special Analysis for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2016-00072). Future SDUs are planned to be 375-foot diameter SDUs, similar to SDU 6 completed in 2017. Construction has already begun on SDU 7 and SDU 8. Excavation has started for SDU 9.

The DOE, through DOE O 435.1, Chg. 1, requires an active maintenance program for the 2009 SDF Performance Assessment (PA), which is satisfied in the *Savannah River Site Liquid Waste Facilities Performance Assessment Maintenance Program FY2019 Implementation Plan.* [SRR-CWDA-2018-00092] A maintenance program is required to continue to reduce uncertainty in the inputs and assumptions in order to provide greater confidence in the results of the analyses and in the long-term plans for public and environmental protection.

The DOE, through DOE O 435.1, Chg. 1, also requires an approved Radioactive Waste Management Basis (RWMB). Potential changes identified during the annual review must be evaluated for impact to the SDF RWMB which consists of facility controls and analyses to demonstrate near- and long-term protection of public, workers, and the environment. Examples of these controls include facility safety documents, waste certification programs, facility waste acceptance requirements, low level waste disposal facility closure plans, PAs, Composite Analyses (CAs), and other facility-specific processes, procedures, and analyses made to comply with DOE O 435.1 and its manual. The current DOE approved SDF RWMB is Q-RMW-Z-00001, Rev. 6. [WDPD-20-04]

The 2009 SDF PA establishes controls to govern waste operations and monitoring performance of the SDF. The mechanisms to demonstrate that operations are within the bounds of the Disposal Authorization Statement (DAS) (WDPD-12-49), the RWMB (Q-RWM-Z-00001), and the PA are Waste Acceptance Criteria (WAC), an Unreviewed Waste Management Question (UWMQ) program, periodic inspections of disposal unit integrity, routine engineering evaluation of inventory and operations, and a comprehensive environmental monitoring program. [X-SD-Z-00004, Manual S4 Procedure ENG.46, SRR-CWDA-2013-00026, Manual SW24.6 Section 2.1, WSRC-TR-2005-00257] Data from these different areas are compared with the critical features, limits, and predictions of the PA to evaluate the performance of the previous fiscal year (FY). The performance evaluation conducted for FY2019 made the following determinations:

- The current performance evaluation conducted on SDU 1, SDU 4, SDU Cells 2A/2B, SDU Cells 3A/3B, SDU Cells 5A/5B, and SDU 6 indicates SDF operations through FY2019 were within the performance expectations of the 2009 SDF PA, the FY2013 SDF Special Analysis (SA), the FY2014 SDF SA, and the FY2016 SDF SA.
- The total inventory of radionuclides accumulated in SDU 1, SDU 4, SDU Cells 2A/2B, SDU Cell 3A, SDU Cells 5A/5B, and SDU 6 through FY2019 was within acceptable

inventory values as defined in the 2009 SDF PA, the FY2013 SDF SA, the FY2014 SDF SA, and the FY2016 SDF SA.

- The one Unreviewed Waste Management Question Evaluation (UWMQE) completed in FY2019 for the SDF was SRR-UWMQE-2017-00003, Revision 2, *Disposal of Tank Closure Cesium Removal DSS at Saltstone Disposal Facility*. SRR-UWMQE-2017-00003 Revision 2 documented disposing decontaminated salt solution (DSS) containing titanium and zirconium leached from ion exchange media from Tank Closure Cesium Removal (TCCR) operations at the SDF does not impact the conclusion of the SDF PA and associated SDF SAs, the CA, or the Waste Determination (WD).
- Research was completed in FY2019 with respect to several ongoing studies on properties considered critical to the performance of saltstone. A more in-depth discussion of on-going and future studies can be found in the *Savannah River Site Liquid Waste Facilities Performance Assessment Maintenance Program FY2019 Implementation Plan*. [SRR-CWDA-2018-00092]
- The routine groundwater monitoring analytical results do not contradict the SDF modeling estimates.

The current performance evaluation conducted on SDU 1, SDU 4, SDU Cells 2A/2B, SDU Cells 3A/3B, SDU Cells 5A/5B, and SDU 6 indicates SDF operations through FY2019 were within the performance expectation of the 2009 SDF PA, the FY2013 SDF SA, FY2014 SDF SA, and FY2016 SDF SA and comply with the DAS, the RWMB, and DOE O 435.1 requirements.

As documented in SRR-CWDA-2017-00078, *FY2017 Annual Review Saltstone Disposal Facility (Z Area) Performance Assessment*, efforts to create a revision to the 2009 SDF PA (SRR-CWDA-2009-00017) were started in FY2017. These efforts have continued through FY2019, with Revision B currently under review by the U.S. Department of Energy Low Level Waste Disposal Facility Federal Review Group (LFRG). The FY2019 SDF PA revision addresses the updated General Separations Area (GSA) database, lessons learned, and new data from research and development activities. The FY2019 SDF PA revision will also incorporate any necessary changes evaluated in the FY2013, FY2014, and FY2016 SA's.

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ACRONYMS

ARP	Actinide Removal Process
BFS	Blast Furnace Slag
CA	Composite Analysis
CFR	U.S. Code of Federal Regulations
CY	Calendar Year
DAS	Disposal Authorization Statement
DLM	Dynamic Leaching Method
DOE	U.S. Department of Energy
DOE-HQ	U.S. Department of Energy – Headquarters
DSA	Documented Safety Analysis
EPA	U.S. Environmental Protection Agency
FA	Fly Ash
FTF	F-Area Tank Farm
FY	Fiscal Year
GGBFS	Ground Granulated Blast Furnace Slag
GSA	General Separations Area
GWPS	Groundwater Protection Standard
HELP	Hydrologic Evaluation of Landfill Performance
HTF	H-Area Tank Farm
K_d	Distribution Coefficient
kCi	Kilocuries
LAZ	Lower Aquifer Zone
LFRG	Low Level Waste Disposal Facility Federal Review Group
LLW	Low-Level Waste
MCU	Modular Caustic Side Solvent Extraction Unit
MF	Monitoring Factor
MOP	Member of the Public
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
NRC	Nuclear Regulatory Commission
NVB	Nonvolatile beta
OM	Organic Matter
OPC	Ordinary Portland Cement
PA	Performance Assessment
PQL	Practical Quantitation Limit
RadFLEx	Radionuclide Field Lysimeter Experiment

RWMB	Radioactive Waste Management Basis
R&D	Research and Development
SA	Special Analysis
SCDHEC	South Carolina Department of Health and Environmental Control
SDF	Saltstone Disposal Facility
SDU	Saltstone Disposal Unit
SHC	Saturated Hydraulic Conductivity
SPF	Saltstone Production Facility
SRNL	Savannah River National Laboratory
SRR	Savannah River Remediation LLC
SRS	Savannah River Site
SWD	Salt Waste Disposal
SWPF	Salt Waste Disposal Facility
TCCR	Tank Closure Cesium Removal
TCCZ	Tan Clay Confining Zone
UAZ	Upper Aquifer Zone
UTRA	Upper Three Runs Aquifer
UWMQ	Unreviewed Waste Management Question
UWMQE	Unreviewed Waste Management Question Evaluation
WAC	Waste Acceptance Criteria
WD	Waste Determination
XRD	X-Ray Diffraction
XRF	X-Ray Fluorescence

1. PURPOSE OF REVIEW

The Saltstone Disposal Facility (SDF) is managed by Savannah River Remediation LLC (SRR) for the U. S. Department of Energy (DOE). The SDF presently consists of Saltstone Disposal Units (SDUs) 1, 2, 3, 4, 5, and 6 as described in the *Performance Assessment for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2009-00017), the *FY2013 Special Analysis for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2013-00062), the *FY2014 Special Analysis for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2014-00006), and the *FY2016 Special Analysis for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2016-00072). Future SDUs are planned to be 375-foot diameter SDUs, similar to SDU 6 completed in 2017. Construction has already begun on SDU 7 and SDU 8. Excavation has started for SDU 9.

The 2009 SDF Performance Assessment (PA) evaluates potential dose impact on a future hypothetical member of the public (MOP), an inadvertent intruder, as well as impacts to the environment from the Low-Level Waste (LLW) disposal facility. [SRR-CWDA-2009-00017] In addition, the 2009 SDF PA demonstrates a reasonable expectation of compliance with pertinent performance objectives as identified in Chapter IV of DOE Manual 435.1-1 and Title 10, of the U.S. Code of Federal Regulations (CFR) Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste,* Subpart C (10 CFR 61) as required by the *Ronald W. Reagan National Defense Authorization Act (NDAA) for Fiscal Year 2005,* Section 3116 (NDAA_3116).

The DOE, through DOE O 435.1, Chg. 1, requires an active maintenance program for the 2009 SDF PA, which is satisfied in the *Savannah River Site Liquid Waste Facilities Performance Assessment Maintenance Program FY2019 Implementation Plan*. [SRR-CWDA-2018-00092] A maintenance program is required to continue to reduce uncertainty in the inputs and assumptions in order to provide greater confidence in the results of the analyses and in the long-term plans for public and environmental protection. Additionally, a disciplined process to address potential changes in disposal operations and/or discoveries (e.g., new waste forms, change in disposal unit design) is required to ensure that proposed changes do not adversely affect SDF performance. Another purpose of the PA maintenance program is to confirm the continued adequacy of the PA through annual reviews of the disposal facility activities. In accordance with the DOE Standard, *Disposal Authorization Statement and Tank Closure Documentation* (DOE-STD-5002-2017), the reviews evaluate and document that SDF operations comply with the Disposal Authorization Statement (DAS) (WDPD-12-49) and DOE O 435.1 requirements and determine if the 2009 SDF PA remains valid or if additional actions are required. A review of the 2009 SDF PA was conducted in a systematic manner that incorporates all the following considerations.

1. Radionuclide inventories, waste volumes, and waste types - The review of waste radionuclide inventories and waste volumes includes a comparison of the actual waste receipts to the evaluated inventory.
2. Research and development (R&D) - The R&D activities are primarily documented in technical reports. The R&D activities are designed and funded to provide additional information for further reduction in the uncertainties associated with PAs results. In addition, Special Analyses (SAs) or Unreviewed Waste Management Question Evaluations (UWMQEs) may be conducted to provide new information or to understand impacts of potential or actual changes to the physical facility, operations, or disposal inventory.

3. PA monitoring - The current monitoring program (SRR-CWDA-2013-00026) includes sampling of the salt waste feed stream in Tank 50 to characterize inventory, evaluation of final waste form composition, periodic sampling of grout raw materials, and monitoring of groundwater to detect changing trends in performance. Physical inspections of structures ensure SDU conditions are consistent with model inputs. Monitoring of system parameters help ensure that the system produces a grout with physical and chemical properties that are consistent with that described in the PA.

The DOE, through DOE O 435.1, Chg. 1, also requires an approved Radioactive Waste Management Basis (RWMB). Potential changes identified during the annual review must be evaluated for impact to the SDF RWMB which consists of facility controls and analyses to demonstrate near- and long-term protection of public, workers, and the environment. Examples of these controls include facility safety documents, waste certification programs, facility waste acceptance requirements, low level waste disposal facility closure plans, PAs, Composite Analyses (CAs), and other facility-specific processes, procedures, and analyses made to comply with DOE O 435.1 and its manual. The current DOE approved SDF RWMB is Q-RMW-Z-00001, Rev. 6. [WDPD-20-04]

All these factors are reviewed annually to evaluate the need to conduct special studies or to prepare a revision of the 2009 SDF PA. Ongoing efforts have prepared a revision to the 2009 SDF PA. The new revision of the SDF PA is currently under review by the Low Level Waste Disposal Facility Federal Review Group (LFRG). It will be implemented in the facility and issued to the public once the review is complete, key issues have been resolved, and the LFRG and DOE-Headquarters (HQ) approves the new DAS.

2. CHANGES POTENTIALLY AFFECTING THE PA, CA, DAS, OR RWMB

2.1 Special Analyses and Unreviewed Waste Management Question Evaluations

2.1.1 Special Analyses

No Special Analyses (SA) were issued during FY2019. The most recent SA, FY2016 SDF SA, was issued in the first quarter of FY2017 (SRR-CWDA-2016-00072).

2.1.2 Unreviewed Waste Management Question Evaluations

One UWMQE was completed in FY2019 for the SDF. The UWMQE was entitled *Disposal of Tank Closure Cesium Removal DSS at Saltstone Disposal Facility* (SRR-UWMQE-2017-00003, Rev.2) and was issued in March 2019. This UWMQE was updated to document the evaluation of disposing decontaminated salt solution (DSS) resulting from Tank Closure Cesium Removal (TCCR) operations at the SDF containing zirconium and titanium leached from the ion exchange media. SRNL-TR-2018-00258, referenced by Revision 2 of the UWMQE, provides an evaluation of the TCCR DSS waste stream and proposed SDF disposal and concludes that disposing of TCCR DSS containing zirconium and titanium leached from the ion exchange media at the SDF are not expected to have an impact on the cured properties of saltstone. The UWMQE states that the proposed activity does not impact the conclusion of the SDF PA, the associated SDF SAs, the Composite Analysis (CA) (SRNL-STI-2009-00512),

or the Waste Determination (WD) (DOE-WD-2005-001). The changes evaluated by the UWMQE indicate that the conclusions in the PA remain valid.

Table 2.1-1: Potential Changes Affecting the PA, CA, DAS, or RWMB

Disposal Facility/Unit	UWMQE	Change, Discovery, Proposed Action, New Information Description	Evaluation Results	Special Analysis Number (if applicable)	PA, CA, or DAS Impacts
SDF	SRR-UWMQE-2017-00003 Rev. 2	Disposal of TCCR DSS containing zirconium and titanium at SDF	The introduction of TCCR DSS containing zirconium and titanium to SDF are not expected to have an impact on the cured properties of saltstone.	N/A	None

2.2 Update the Closure Plan

Management of SRS LLW is regulated under DOE M 435.1-1, *Radioactive Waste Management Manual*. A DAS (WDPD-12-49) revision was issued by DOE on May 22, 2012 authorizing continued operations of the SDF. The DAS specifies the closure plan that complies with DOE M 435.1-1 for SDF must be maintained and modified as needed to reflect facility changes. The SDF closure plan is reviewed annually to determine if a revision is required. The Closure Plan was updated in FY2015. An update to the Closure Plan will be prepared upon approval of the updated FY2019 SDF PA.

The *Savannah River Site Land Use Plan* (SRNS-RP-2014-00537) provides the framework for integrating the SRS mission and vision with ecological, economic, cultural, and social factors in a regional context and to support decision-making for near-term and long-term use of the site, including the SDF. The *Land Use Plan* describes the current site conditions, defines a vision for the evolution of the site, outlines actions to achieve the vision, and guides the allocation of resources toward attainment of that vision. This plan provides guidance and direction for the future physical development of the site and provides a framework within which detailed analyses will be conducted to determine the courses of action required to reach optimum site configuration. The plan is based on specific assumptions. If these assumptions were to change, the plan would be updated to reflect the changed conditions. The *Land Use Plan* was issued in November 2014 and no modifications have occurred to the plan, therefore there are no impacts to the current SDF PA/SA.

2.3 Update the SDF Monitoring Plan

The *Performance Assessment Monitoring Plan for the Saltstone Disposal Facility at the Savannah River Site* (SRR-CWDA-2013-00026) (hereinafter referred to as the SDF Monitoring Plan), demonstrates compliance with pertinent requirements of DOE O 435.1, Chg. 1 and its associated Manual and Guide. The SDF Monitoring Plan was issued in FY2013 based on the 2009 SDF PA to incorporate ongoing activities as required by the DAS. [WDPD-12-49]

The SDF Monitoring Plan that complies with DOE M 435.1-1 must be maintained and modified as needed to reflect facility changes. The Monitoring Plan is reviewed annually to determine if a

revision is required. Revision 1 to the SDF Monitoring Plan was completed in August 2015. [SRR-CWDA-2013-00026] The revision incorporated and integrated the ongoing activities relative to new groundwater wells and the 375-foot diameter SDU design at the SDF. An update to the Monitoring Plan will be prepared upon approval of the FY2019 SDF PA.

3. CUMULATIVE EFFECTS OF CHANGES

One UWMQE was performed in FY2019 (SRR-UWMQE-2017-00003, Rev. 2). The UWMQE evaluated the impacts of the TCCR DSS stream containing zirconium and titanium leached from the ion exchange media to the SDF and found no impacts on the cured properties of saltstone. Therefore, there was no impact to the conclusion of the SDF PA, the associated SDF SAs, the CA, the RWMB, or the WD. An SDF PA revision is in progress with updates to the General Separations Area (GSA) Model inputs and new data from ongoing R&D activities. FY2019 SDF PA Revision B is currently under review by the LFRG.

4. WASTE RECEIPTS

4.1 Waste Volumes and Radionuclide Inventories

Construction of rectangular shaped SDUs 1 and 4 was completed between February 1986 and July 1988. The Saltstone Production Facility (SPF) started radioactive operations June 1990. Disposal into SDU 1 occurred intermittently from June 1990 to September 1996. Disposal into SDU 4 began in January 1997 and completed in FY2012.

Initially constructed cylindrical SDUs (2A/2B, 3A/3B and 5A/5B) include six cells, 150 feet in diameter by 22 feet high. Future SDUs are anticipated to consist of single cells, 375 feet in diameter and 45 feet high, such as SDU 6 which was placed into operation in 2017. Figure 4.1-1 shows the October 2019 configuration of SDUs at the SDF. Figure 4.1-1 shows the ongoing construction efforts for SDU 7 and SDU 8 as well as the beginning excavation for SDU 9 (SRR-CWDA-2020-00010). Disposal into SDU Cell 2B commenced in September 2012 and disposal into SDU Cell 2A commenced in December 2012. SDU Cells 2A and 2B have a slightly less nominal useable volume for grout disposal due to an additional one foot of clean pour (Class III sulfate resistant concrete) having been placed on the floor. The pour was performed to address suspected compromised areas detected during water tightness testing. Disposal into SDU Cell 2A concluded in June 2014 and disposal into SDU Cell 2B concluded in July 2014. Disposal into SDU Cell 5B commenced in August 2014 and concluded in February 2017 and disposal into SDU Cell 5A commenced in August 2015 and concluded in August 2016. Disposal into SDU Cell 3A commenced in February 2017; however, there was no disposal into SDU Cell 3A during FY2019. The first disposal into SDU 6 was in August 2018 and continued through FY2019. No disposal into SDU Cell 3B has occurred. Table 4.1-1 summarizes the SDU waste receipts (i.e., saltstone grout emplaced in each SDU) through FY2019. Table 4.1-1 also provides both the PA-estimated disposal capacity and the operational capacity for each SDU. The PA-estimated disposal capacity is the assumed grout disposal capacity for the SDU used in the PA; while the operational capacity is the actual grout disposal capacity based on operational limits imposed by the Saltstone Facility Documented Safety Analysis (DSA) (WSRC-SA-2003-00001).

Future SDUs will be constructed, as needed, in coordination with salt processing production rates. The anticipated quantity and need dates for future SDUs are outlined in the *Liquid Waste System*

Plan. [SRR-LWP-2009-00001] The *Liquid Waste System Plan* is updated as necessary to align with the most recent operational, budgetary, and regulatory requirements of the overall Liquid Waste System at the Savannah River Site (SRS). The most recent revision to the plan is Revision 21.

Figure 4.1-1: Saltstone Facility Aerial View (October 2019)

[SRR-CWDA-2020-00010]

Table 4.1-1: Saltstone Disposal Unit Waste Receipts Through FY2019

Saltstone Disposal Unit	Disposal Volume (gal) to date	PA-Estimated Disposal Capacity (gal)[a]	Operational Capacity (gal)[b,c,d, e]	Percent Filled (%) Volume[f]	Total Curies Disposed to date (kCi)	PA/CA Impacts
1	5,610,000	10,900,000	10,900,000	51	166	None
2	5,540,000	5,660,000	5,600,000	100	30.2	None
3	693,000	5,660,000	5,600,000	12	2.09	None
4	19,070,000	21,800,000	20,000,000	95	508[g]	None
5	5,540,000	5,660,000	5,600,000	100	26.0	None
6	1,295,000[h,i]	33,700,000	32,800,000	3.9	4.66	None

Note: The "Sum of Fractions or Total curie vs PA Curie Limit" column outlined in the DOE Standard (DOE-STD-5002-2017) is not presented in this table as the individual SDUs do not have a PA curie limit placed on them.

a. PA-estimated disposal capacities are from SRR-CWDA-2017-00032, Table 2.3-1, SDU Fillable Volume.
b. Operational capacity for SDU 1 is the same as the PA-Estimated Disposal Capacity for SDU 1.
c. Operational capacity for SDU 4 is based on an SDU cell fill height of 22.5 feet.
d. Operational capacity for SDU's 2, 3, and 5 are based on an SDU cell fill height of 21.25 feet (X-CLC-Z-00070, X-CLC-Z-00078, X-CLC-Z-00080).
e. Operational capacity for SDU 6 is based on SDU 6 fill height limit of 41 feet (SRR-SDU-2017-00003).
f. Percent filled volume is based on comparison of disposal volume to date to operational capacity.
g. SDU 4 inventory includes the inventory of 10,032 United States Naval Fuel Material Facility 55-gallon drums emplaced in Cell A (SRR-CWDA-2018-00072).
h. The volume of saltstone grout emplaced in SDU 6 is based on the ratio of salt solution to grout (0.633) per the methodology in X-CLC-Z-00086.
i. The volume of saltstone grout emplaced in SDU 6 does not include the minimal volume of non-radioactive grout (i.e., clean cap) that was placed into SDU 6 during start-up testing. [X-CLC-Z-00085]

During FY2019, no additional saltstone was disposed in SDU 1, SDU 4, SDU Cells 2A/2B, SDU Cells 3A/3B, or SDU Cells 5A/5B, therefore the inventories for these SDUs are presented in this annual review to reflect only current decayed inventories (i.e., end of FY2019 (September 30, 2019)), and decayed inventories at closure (i.e., January 1, 2032). The inventory for SDU 6 was updated to reflect the disposal operations over the past year. The inventories for SDU 3A and SDU 6 were also updated to reflect a comparison of decayed inventories at closure to those modeled in the FY2014 SDF SA and FY2016 SDF SA. These updates are documented in SRR-CWDA-2018-00072, *Determination of SDF Inventories through 9/30/2018*, Savannah River Site, Aiken, SC, Rev. 0, December 2018.

4.1.1 Waste Volumes

The salt solution production history through FY2019 is presented in Table 4.1-2, and new waste receipts into the SDF in FY2019 are detailed in Table 4.1-3. An estimate of the remaining grout capacity of SDUs 1 and 4 can be made by comparing the total grout capacity to the level in each cell times the cell surface area (see Table 4.1-1).

SDU 1 is currently at 51 % of its operational capacity with Cells A, B, and C being full. SDU 1 Cells D, E, and F are empty. Currently, there are no plans to use SDU 1 Cells D, E, and F for saltstone disposal. [SRR-LWP-2009-00001] Therefore, no additional waste receipts were placed in SDU 1 during FY2019.

SDU Cell 2A has reached 100 % of its operational capacity (21.25 ft of height used relative to 21.25 ft of height available). [X-CLC-Z-00070] SDU Cell 2B has reached 100 % of its operational capacity (21.25 ft of height used relative to 21.25 ft of height available). [X-CLC-Z-00070]

SDU Cell 3A has reached 25 % of its operational capacity (5.25 ft of height used relative to 21.25 ft of height available). [X-CLC-Z-00085] No saltstone grout has been emplaced in SDU Cell 3B as of the end of FY2019. Table 4.1-1 reflects a 12 % filled volume based on comparing the SDU Cell 3A used volume to the combined SDU Cells 3A and 3B operational capacities.

SDU 4 has reached 95% of its operational capacity (930,000 gallons of remaining disposal volume of the 20-million-gallon capacity), not including final clean cap installation. [X-CLC-Z-00052] No additional saltstone was disposed in SDU 4 during FY2019 and there are no plans to dispose of additional saltstone in SDU 4 in the future. SDU 4 Cell A contains 10,032 drums (added to SDU 4 in the 1990s) of a non-hazardous cementitious waste form (referred to as saltcrete) generated from the United States Naval Fuel Material Facility wastewater treatment operation. SDU 4 Cell A also contains the wooden pallets used in the movement of the drums (one pallet to four drums). [ESH-FSS-9000373] The void space surrounding the drums in SDU 4 Cell A is filled with clean grout.

SDU Cell 5A has reached 100 % of its operational capacity (21.25 ft of height used relative to 21.25 ft of height available). [X-CLC-Z-00078] SDU Cell 5B has reached 100 % of its operational capacity (21.25 ft of height used relative to 21.25 ft of height available). [X-CLC-Z-00080]

SDU 6 has reached 3.9 % of its operational capacity. The percent filled volume is based on a disposal volume of 1,295,000 gallons of saltstone grout emplaced into SDU 6 by the end of FY2019 [SRR-CWDA-2019-00110]. The methodology used to determine the volume of saltstone grout disposed is provided in Notes h and i of Table 4.1-1, as well as in SRR-CWDA-2018-00072.

The FY2019 performance evaluation, conducted on SDU 1, SDU Cells 2A/2B, SDU Cell 3A, SDU 4, SDU Cells 5A/5B and SDU 6, indicates SDF operations were within the performance expectation of the 2009 SDF PA, the FY2013 SDF SA, the FY2014 SDF SA, and the FY2016 SDF SA. [SRR-CWDA-2009-00017, SRR-CWDA-2013-00062, SRR-CWDA-2014-00006, SRR-CWDA-2016-00109] All SDF disposal operations are limited by the *Waste Acceptance Criteria for Transfers to the Z-Area Saltstone Production Facility During Salt Disposition Integration (SDI)* (X-SD-Z-00004).

Table 4.1-2: FY2019 Tank 50 Salt Solution Historical Data

Fiscal Year	Salt Solution Processed (gal)					
	SDU 1	SDU 4	SDU 2	SDU 3	SDU 5	SDU 6
1990	246,660	0	0	0	0	0
1991	651,279	0	0	0	0	0
1992	105,391	0	0	0	0	0
1993	28,020	0	0	0	0	0
1994	261,058	0	0	0	0	0
1995	129,900	0	0	0	0	0
1996	607,774	0	0	0	0	0
1997	0	212,370	0	0	0	0
1998	0	339,310	0	0	0	0
1999	0	0	0	0	0	0
2000	0	0	0	0	0	0
2001	0	0	0	0	0	0
2002	0	263,830	0	0	0	0
2003	0	1,292,474	0	0	0	0
2004	0	0	0	0	0	0
2005	0	0	0	0	0	0
2006	0	0	0	0	0	0
2007	0	244,480	0	0	0	0
2008	0	1,342,930	0	0	0	0
2009	0	1,525,728	0	0	0	0
2010	0	1,013,770	0	0	0	0
2011	0	1,486,842	0	0	0	0
2012	0	811,710	439,740	0	0	0
2013	0	0	2,005,340	0	0	0
2014	0	0	486,474	0	680,146	0
2015	0	0	0	0	828,128	0
2016	0	0	0	0	1,506,010	0
2017	0	0	0	61,600	108,060	0
2018	0	0	0	268,384	0	116,875
2019	0	0	0	0	0	703,300
SDU Totals:	2,030,082	8,533,444[a]	2,931,554	329,984	3,122,344	820,175
Total Salt Solution Processed Through End of September 2019:	17,767,583					

[X-CLC-Z-00084, X-CLC-Z-00085, X-CLC-Z-00086, SRR-CWDA-2018-00072, SRR-CWDA-2019-00110]

a. This volume is only the gallons of Tank 50 salt solution processed to SDU 4. The volume presented does not include the 10,032 United States Naval Fuel Material Facility 55-gallon drums emplaced in Cell A as the waste was not from Tank 50 salt solution processing.

Table 4.1-3: Tank 50 Salt Solution Processed

Time Period	Salt Solution Processed (gal)	SDU
1st Quarter FY2019	142,153	SDU 6
2nd Quarter FY2019	466,606	SDU 6
3rd Quarter FY2019	94,541	SDU 6
4th Quarter FY2019	0	SDU 6
Total FY2019 Receipts	703,300	SDU 6

[X-CLC-Z-00086, X-CLC-Z-00087, X-CLC-Z-00088, X-CLC-Z-00089]

4.1.2 Waste Inventory

A disposed radionuclide inventory estimate was developed for use in PA modeling. [SRNS-J2100-2008-00004] The 2009 SDF PA, based on these inventory estimates, met performance objectives. SDF inventories were updated for preparation of the FY2014 SDF SA and documented in *SDF Inventory Estimates for Transport Modeling* (SRR-CWDA-2013-00147). The updated inventories developed for the FY2014 SDF SA were then used for the FY2016 SDF SA.

Determination of SDF Inventories through 9/30/2019 (SRR-CWDA-2019-00110) includes both the original inventory disposed of at the SDF and the current inventory through FY2019. The current inventory includes decay and ingrowth for SDF operations beginning in 1990 through FY2019. As of the end of FY2019, 738 kilocuries (kCi) have been disposed in the SDF and the current inventory as of the end of FY2019, accounting for decay and daughter ingrowth, is 383 kCi. [SRR-CWDA-2019-00110]

In FY2019, 703 kgal of low-level waste was transferred from Tank 50 to the SPF. A total of 1,110 kgal of saltstone was emplaced in the SDF containing 3.92 kCi. All saltstone in FY2019 was emplaced in SDU 6. [SRR-CWDA-2019-00110]

The current inventories were decayed to 10/1/2032 to match the basis date for closure used in the FY2014 and FY2016 SDF SAs. For SDU 1 and 4, as well as SDU Cells 2A/2B and SDU Cells 5A/5B, no additional saltstone was disposed in FY2019 and there are no plans to place additional saltstone in these units in the future; therefore, Tables 4.1-4 through 4.1-9 provide only current inventories as of 9/30/2019 and inventories decayed to 10/1/2032.

For SDU Cell 3A, no saltstone was disposed in FY2019. However, it will be used for future processing. So, for SDU Cell 3A (Table 4.1-10) and for SDU 6 (Table 4.1-11), the current FY2019 inventory (decayed to 10/1/2032) was divided by the FY2014 SDF SA inventory to generate a ratio. This ratio helps in evaluating the inventory that will be in each SDU upon completion of filling.

The total inventory of radionuclides in SDU 1, SDU 4, SDU Cells 2A/2B, and SDU Cells 5A/5B through FY2019 was below that shown to be acceptable and do not contradict the performance objectives of the 2009 SDF PA, the FY2013 SDF SA, the FY2014 SDF SA, or the FY2016 SDF SA.

The comparison ratios presented in Table 4.1-10 indicate that the current SDU 3A inventories for three radionuclides (Cs-135, I-129, and Nb-94) are currently trending higher than would be expected for SDU 3A relative to the inventory values that were evaluated in the FY2014 SDF

SA (SRR-CWDA-2014-00006). Given that 25% of the SDU 3A volume has been filled (indicated as 12% filled in Table 4.1-1 for both 3A and 3B together), the comparison ratios were not expected to exceed 2.50E-01. For Cs-135 and Nb-94, the inventory estimates are based on detection limit values and/or special methods which are likely to be over-predicting the concentrations. Additionally, recent salt batches have had above-average concentrations of I-129 (e.g., compare I-129 concentration from SRNL-STI-2015-00622 versus SRNL-STI-2013-00437). Regardless, these values fall within the ranges of uncertainty that were considered within the FY2014 SDF SA (SRR-CWDA-2014-00006). Further, based on more recent analytical approaches, the current inventory estimates are only trending slightly higher than more recent estimates (SRR-CWDA-2018-00044). Finally, because SDU 3A is significantly smaller than the 375-foot diameter SDUs and because it is located above a subsurface ground water divide, it is unlikely to become a significant contributor to the total dose results. Therefore, these higher-than-expected concentrations of Cs-135, I-129, and Nb-94 within SDU 3A are not expected to be risk-significant.

Similarly, the comparison ratios presented in Table 4.1-11 indicate that the current SDU 6 inventories for these same radionuclides (Cs-135, I-129, and Nb-94), as well as U-232, are also currently trending higher than would be expected for SDU 6 relative to the inventory values that were evaluated in the FY2014 SDF SA (SRR-CWDA-2014-00006). Given that 3.9% of the SDU 6 volume has been filled, the comparison ratios were not expected to exceed 3.90E-02. The inventories of Pu-242, Pu-244, U-234 are also slightly higher than this 3.90E-02 ratio threshold. Some variability is to be expected in the waste concentrations during disposal operations, and with more than 96% of the SDU volume still available, it is too early to draw conclusions based on the current values. Nevertheless, because SDU 6 is further from the 100-meter boundary than other SDUs (see Figure 5.2-1 and Table 5.2-2 in SRR-CWDA-2014-00006), slightly higher concentrations in SDU 6 are unlikely to significantly affect the long-term performance of the SDF, because any potential releases from SDU 6 will undergo plume spreading before reaching the points of assessment along the 100-meter boundary.

SRR will continue to monitor the inventories in each SDU and evaluate if the results are within the uncertainty values modeled in the SDF PA. When each SDU is filled it will be evaluated to determine if the inventories are within PA limits and, if exceeded, a UWMQE, or other appropriate evaluation, may be performed to determine impacts.

Table 4.1-4: Saltstone Disposal Facility SDU 1 Inventory

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Ac-227	1.74E-06	2.35E-06
Al-26	2.62E-01	2.62E-01
Am-241	2.01E-03	2.26E-03
Am-242m	6.75E-05	6.33E-05
Am-243	1.42E-03	1.42E-03
Ba-137m[b]	5.72E+00	4.24E+00
C-14	1.31E+00	1.31E+00
Cf-249	8.76E-13	8.54E-13
Cf-251	3.08E-14	3.05E-14
Cl-36	9.75E-08	9.75E-08
Cm-243	4.43E-04	3.25E-04
Cm-244	2.80E-03	1.70E-03
Cm-245	2.72E-04	2.71E-04
Cm-247	1.59E-13	1.59E-13
Cm-248	1.66E-13	1.66E-13
Co-60	3.38E-04	6.10E-05
Cs-135	4.95E-02	4.95E-02
Cs-137	6.06E+00	4.49E+00
Eu-152	1.40E-03	7.21E-04
Eu-154	4.53E-04	1.59E-04
H-3	1.22E+01	5.85E+00
I-129	2.01E-01	2.01E-01
K-40	9.75E-08	9.75E-08
Nb-93m	7.55E-01	7.53E-01
Nb-94	2.03E-03	2.03E-03
Ni-59	2.30E-03	2.30E-03
Ni-63	1.18E-01	1.08E-01
Np-237	3.94E-03	3.94E-03
Pa-231	3.16E-06	3.85E-06
Pd-107	8.38E-03	8.38E-03
Pt-193	1.61E+00	1.34E+00
Pu-238	7.08E-03	6.39E-03
Pu-239	1.43E-02	1.43E-02
Pu-240	1.35E-02	1.35E-02
Pu-241	1.86E-02	1.01E-02
Pu-242	1.57E-03	1.57E-03
Pu-244	1.01E-05	1.01E-05
Ra-226	5.33E-07	8.39E-07
Ra-228	7.68E-06	7.68E-06
Se-79	3.44E-01	3.44E-01
Sm-151	5.46E-03	4.94E-03

Table 4.1-4: Saltstone Disposal Facility SDU 1 Inventory (Continued)

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032[c] (Ci)
Sn-126	1.22E+00	1.22E+00
Sr-90	1.47E-02	1.08E-02
Tc-99	4.93E+01	4.93E+01
Th-229	4.17E-04	5.05E-04
Th-230	4.91E-05	6.09E-05
Th-232	7.69E-06	7.69E-06
U-232	6.42E-04	5.63E-04
U-233	7.76E-02	7.76E-02
U-234	9.93E-02	9.93E-02
U-235	2.51E-03	2.51E-03
U-236	6.49E-03	6.49E-03
U-238	1.07E-02	1.07E-02
Y-90[b]	1.47E-02	1.08E-02
Zr-93	7.69E-01	7.69E-01

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[c] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

Table 4.1-5: Saltstone Disposal Facility SDU 4 Inventory

Radionuclide	Current Inventory[a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Ac-227	6.53E-05	8.79E-05
Al-26	9.75E-01	9.75E-01
Am-241	2.07E+01	2.15E+01
Am-242m	1.89E-02	1.77E-02
Am-243	5.18E-01	5.17E-01
Ba-137m[b]	1.54E+05	1.14E+05
C-14	6.51E+00	6.50E+00
Cf-249	2.76E-01	2.69E-01
Cf-251	9.29E-02	9.20E-02
Cl-36	1.49E-02	1.49E-02
Cm-243	7.96E-03	5.84E-03
Cm-244	3.02E+01	1.83E+01
Cm-245	7.78E-01	7.78E-01
Cm-247	1.06E-01	1.06E-01
Cm-248	7.43E-13	7.43E-13
Co-60	5.53E-02	9.96E-03
Cs-135	1.73E+00	1.73E+00
Cs-137	1.63E+05	1.21E+05
Eu-152	6.69E-02	3.44E-02
Eu-154	2.70E+00	9.45E-01
H-3	3.23E+01	1.56E+01
I-129	2.77E-01	2.77E-01
K-40	1.49E-02	1.49E-02
Nb-93m	1.00E+03	1.76E+03
Nb-94	8.93E-02	8.93E-02
Ni-59	7.89E-02	7.89E-02
Ni-63	3.09E+00	2.83E+00
Np-237	5.76E-01	5.76E-01
Pa-231	1.18E-04	1.44E-04
Pd-107	3.75E-02	3.75E-02
Pt-193	8.68E+00	7.25E+00
Pu-238	3.08E+02	2.78E+02
Pu-239	5.86E+01	5.86E+01
Pu-240	7.28E+01	7.27E+01
Pu-241	8.21E+01	4.41E+01
Pu-242	4.12E+00	4.12E+00
Pu-244	1.68E-02	1.68E-02
Ra-226	4.12E-05	7.15E-05
Ra-228	2.10E-04	2.10E-04
Se-79	9.75E+00	9.75E+00
Sm-151	1.94E+01	1.76E+01

Table 4.1-5: Saltstone Disposal Facility SDU 4 Inventory (Continued)

Radionuclide	Current Inventory[a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Sn-126	2.22E+00	2.22E+00
Sr-90	2.27E+03	1.66E+03
Tc-99	6.34E+02	6.34E+02
Th-229	3.63E+00	3.64E+00
Th-230	4.90E-03	5.97E-03
Th-232	2.10E-04	2.10E-04
U-232	1.16E-01	1.02E-01
U-233	8.86E+00	8.86E+00
U-234	8.97E+00	8.98E+00
U-235	9.37E-02	9.37E-02
U-236	8.34E-02	8.35E-02
U-238	7.93E-02	7.93E-02
Y-90[b]	2.27E+03	1.66E+03
Zr-93	8.15E+00	8.15E+00

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[c] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

Table 4.1-6: Saltstone Disposal Facility SDU Cell 2A Inventory

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Ac-227	6.94E-07	9.32E-07
Al-26	8.98E-04	8.98E-04
Am-241	6.44E-02	8.64E-02
Am-242m	9.89E-03	9.28E-03
Am-243	4.21E-03	4.22E-03
Ba-137m[b]	5.60E+03	4.15E+03
C-14	2.41E+00	2.41E+00
Cf-249	1.78E-02	1.73E-02
Cf-251	1.34E-02	1.32E-02
Cl-36	1.06E-04	1.06E-04
Cm-243	5.15E-04	3.78E-04
Cm-244	1.83E-01	1.11E-01
Cm-245	1.45E-02	1.46E-02
Cm-247	1.67E-02	1.67E-02
Cm-248	1.20E-13	1.20E-13
Co-60	7.22E-04	1.30E-04
Cs-135	3.36E-02	3.36E-02
Cs-137	5.93E+03	4.40E+03
Eu-152	2.36E-04	1.21E-04
Eu-154	7.12E-03	2.49E-03
H-3	2.10E+00	1.01E+00
I-129	7.31E-02	7.31E-02
K-40	1.06E-04	1.06E-04
Nb-93m	2.77E-01	2.69E-01
Nb-94	1.90E-03	1.89E-03
Ni-59	9.16E-04	9.16E-04
Ni-63	4.39E-02	4.01E-02
Np-237	1.60E-01	1.60E-01
Pa-231	1.25E-06	1.53E-06
Pd-107	6.03E-03	6.03E-03
Pt-193	1.53E+00	1.28E+00
Pu-238	5.63E+00	5.08E+00
Pu-239	5.28E-01	5.28E-01
Pu-240	5.28E-01	5.27E-01
Pu-241	1.53E+00	8.20E-01
Pu-242	3.76E-01	3.76E-01
Pu-244	1.74E-03	1.74E-03
Ra-226	2.23E-06	6.93E-06
Ra-228	1.26E-05	1.26E-05
Se-79	1.34E-01	1.34E-01
Sm-151	1.64E-01	1.48E-01

Table 4.1-6: Saltstone Disposal Facility SDU Cell 2A Inventory (Continued)

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Sn-126	7.62E-01	7.62E-01
Sr-90	1.60E+01	1.17E+01
Tc-99	1.14E+02	1.14E+02
Th-229	2.85E-03	3.93E-03
Th-230	8.03E-04	8.76E-04
Th-232	1.26E-05	1.26E-05
U-232	1.35E-02	1.18E-02
U-233	9.51E-01	9.51E-01
U-234	6.16E-01	6.16E-01
U-235	9.93E-04	9.93E-04
U-236	6.37E-03	6.37E-03
U-238	2.24E-02	2.24E-02
Y-90[b]	1.60E+01	1.17E+01
Zr-93	2.64E-01	2.64E-01

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[c] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

Table 4.1-7: Saltstone Disposal Facility SDU Cell 2B Inventory

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Ac-227	8.89E-07	1.20E-06
Al-26	8.56E-04	8.56E-04
Am-241	7.83E-02	9.87E-02
Am-242m	6.29E-03	5.90E-03
Am-243	4.88E-03	4.90E-03
Ba-137m[b]	6.89E+03	5.11E+03
C-14	2.44E+00	2.44E+00
Cf-249	1.46E-02	1.43E-02
Cf-251	1.08E-02	1.07E-02
Cl-36	1.30E-04	1.30E-04
Cm-243	5.12E-04	3.76E-04
Cm-244	2.08E-01	1.26E-01
Cm-245	2.22E-02	2.22E-02
Cm-247	1.41E-02	1.41E-02
Cm-248	1.20E-13	1.20E-13
Co-60	8.08E-04	1.46E-04
Cs-135	3.38E-02	3.38E-02
Cs-137	7.30E+03	5.41E+03
Eu-152	2.37E-04	1.22E-04
Eu-154	1.28E-02	4.47E-03
H-3	1.78E+00	8.57E-01
I-129	6.83E-02	6.83E-02
K-40	1.30E-04	1.30E-04
Nb-93m	4.20E-01	4.00E-01
Nb-94	1.63E-03	1.63E-03
Ni-59	7.32E-04	7.32E-04
Ni-63	3.50E-02	3.20E-02
Np-237	9.61E-02	9.61E-02
Pa-231	1.60E-06	1.96E-06
Pd-107	6.06E-03	6.06E-03
Pt-193	1.53E+00	1.28E+00
Pu-238	5.35E+00	4.83E+00
Pu-239	5.19E-01	5.19E-01
Pu-240	5.19E-01	5.18E-01
Pu-241	1.44E+00	7.79E-01
Pu-242	5.21E-01	5.21E-01
Pu-244	2.42E-03	2.42E-03
Ra-226	1.31E-06	3.94E-06
Ra-228	1.92E-05	1.92E-05
Se-79	1.23E-01	1.23E-01
Sm-151	1.44E-01	1.30E-01

Table 4.1-7: Saltstone Disposal Facility SDU Cell 2B Inventory (Continued)

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Sn-126	6.83E-01	6.83E-01
Sr-90	1.97E+01	1.44E+01
Tc-99	1.37E+02	1.37E+02
Th-229	6.26E-03	7.75E-03
Th-230	4.20E-04	5.22E-04
Th-232	1.92E-05	1.92E-05
U-232	1.31E-02	1.15E-02
U-233	1.32E+00	1.32E+00
U-234	8.54E-01	8.54E-01
U-235	1.27E-03	1.27E-03
U-236	8.85E-03	8.85E-03
U-238	2.65E-02	2.65E-02
Y-90[b]	1.97E+01	1.44E+01
Zr-93	3.83E-01	3.83E-01

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[c] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

Table 4.1-8: Saltstone Disposal Facility SDU Cell 5A Inventory

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Ac-227	4.13E-05	2.80E-05
Al-26	1.02E-03	1.02E-03
Am-241	1.64E-01	5.29E-01
Am-242m	9.99E-04	9.37E-04
Am-243	3.91E-03	3.93E-03
Ba-137m[b]	3.47E+03	2.57E+03
C-14	3.18E+00	3.17E+00
Cf-249	2.22E-02	2.17E-02
Cf-251	1.68E-02	1.66E-02
Cl-36	3.10E-03	3.10E-03
Cm-243	5.90E-04	4.33E-04
Cm-244	9.05E-02	5.50E-02
Cm-245	1.50E-02	1.50E-02
Cm-247	2.13E-02	2.13E-02
Cm-248	1.28E-13	1.28E-13
Co-60	1.90E-02	3.43E-03
Cs-135	3.60E-02	3.60E-02
Cs-137	3.67E+03	2.72E+03
Eu-152	2.53E-04	1.30E-04
Eu-154	3.68E-03	1.29E-03
H-3	5.12E+00	2.46E+00
I-129	1.39E-01	1.39E-01
K-40	3.10E-03	3.10E-03
Nb-93m	5.23E-01	4.26E-01
Nb-94	2.66E-03	2.65E-03
Ni-59	1.26E-03	1.26E-03
Ni-63	6.14E-02	5.61E-02
Np-237	8.51E-02	8.51E-02
Pa-231	1.93E-06	2.35E-06
Pd-107	6.46E-03	6.46E-03
Pt-193	1.70E+00	1.42E+00
Pu-238	7.47E+01	6.75E+01
Pu-239	1.86E+00	1.86E+00
Pu-240	1.86E+00	1.85E+00
Pu-241	2.42E+01	1.29E+01
Pu-242	4.61E-01	4.61E-01
Pu-244	2.14E-03	2.14E-03
Ra-226	4.19E-07	1.75E-06
Ra-228	1.35E-05	1.35E-05
Se-79	2.72E-01	2.72E-01
Sm-151	2.54E-01	2.30E-01

Table 4.1-8: Saltstone Disposal Facility SDU Cell 5A Inventory (Continued)

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Sn-126	1.65E+00	1.65E+00
Sr-90	4.70E+02	3.44E+02
Tc-99	1.75E+02	1.75E+02
Th-229	5.33E-04	1.86E-03
Th-230	1.93E-04	2.83E-04
Th-232	1.35E-05	1.35E-05
U-232	1.42E-02	1.24E-02
U-233	1.17E+00	1.17E+00
U-234	7.56E-01	7.59E-01
U-235	1.53E-03	1.53E-03
U-236	7.82E-03	7.82E-03
U-238	3.28E-02	3.28E-02
Y-90[b]	4.70E+02	3.44E+02
Zr-93	3.05E-01	3.05E-01

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[c] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

Table 4.1-9: Saltstone Disposal Facility SDU Cell 5B Inventory

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Ac-227	6.83E-05	4.58E-05
Al-26	8.59E-04	8.59E-04
Am-241	6.56E-02	1.86E-01
Am-242m	8.24E-04	7.73E-04
Am-243	9.21E-03	9.22E-03
Ba-137m[c]	7.16E+03	5.31E+03
C-14	3.78E+00	3.77E+00
Cf-249	1.83E-02	1.78E-02
Cf-251	1.51E-02	1.50E-02
Cl-36	7.71E-04	7.71E-04
Cm-243	5.73E-04	4.20E-04
Cm-244	1.03E-01	6.24E-02
Cm-245	1.31E-02	1.31E-02
Cm-247	1.77E-02	1.77E-02
Cm-248	1.27E-13	1.27E-13
Co-60	8.78E-04	1.58E-04
Cs-135	3.58E-02	3.58E-02
Cs-137	7.59E+03	5.62E+03
Eu-152	2.52E-04	1.29E-04
Eu-154	3.84E-03	1.34E-03
H-3	3.14E+00	1.51E+00
I-129	8.68E-02	8.68E-02
K-40	1.72E-03	1.72E-03
Nb-93m	3.13E-01	2.98E-01
Nb-94	2.06E-03	2.06E-03
Ni-59	5.15E-04	5.15E-04
Ni-63	2.50E-02	2.29E-02
Np-237	7.76E-02	7.76E-02
Pa-231	1.60E-06	1.95E-06
Pd-107	6.42E-03	6.42E-03
Pt-193	1.67E+00	1.39E+00
Pu-238	2.58E+01	2.33E+01
Pu-239	7.39E-01	7.39E-01
Pu-240	7.39E-01	7.38E-01
Pu-241	8.00E+00	4.27E+00
Pu-242	4.21E-01	4.21E-01
Pu-244	1.95E-03	1.95E-03
Ra-226	4.45E-07	1.67E-06
Ra-228	2.26E-05	2.26E-05
Se-79	1.68E-01	1.68E-01
Sm-151	2.10E-01	1.90E-01

Table 4.1-9: Saltstone Disposal Facility SDU Cell 5B Inventory (Continued)

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [c] (Ci)
Sn-126	9.47E-01	9.47E-01
Sr-90	1.17E+02	8.56E+01
Tc-99	1.20E+02	1.20E+02
Th-229	4.99E-04	1.71E-03
Th-230	1.78E-04	2.60E-04
Th-232	2.26E-05	2.26E-05
U-232	1.94E-02	1.70E-02
U-233	1.07E+00	1.07E+00
U-234	6.87E-01	6.88E-01
U-235	1.27E-03	1.27E-03
U-236	7.18E-03	7.18E-03
U-238	2.85E-02	2.85E-02
Y-90[b]	1.17E+02	8.57E+01
Zr-93	2.86E-01	2.86E-01

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[c] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

Table 4.1-10: Saltstone Disposal Facility SDU Cell 3A Inventory Comparison

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [d] (Ci)	Evaluated Total Inventory per SDU[b] 10/1/2032 (Ci)	Comparison Ratio
Ac-227	3.08E-05	2.05E-05	8.40E-02	2.44E-04
Al-26	1.32E-04	1.32E-04	2.40E+01	5.49E-06
Am-241	4.40E-02	1.97E-01	1.40E+04	1.41E-05
Am-242m	5.36E-05	5.03E-05	9.10E+00	5.53E-06
Am-243	3.77E-04	3.78E-04	6.90E-02	5.48E-03
Ba-137m[c]	8.90E+02	6.59E+02	N/A	N/A
C-14	7.02E-01	7.01E-01	5.80E+02	1.21E-03
Cf-249	8.79E-04	8.57E-04	N/A	N/A
Cf-251	6.09E-04	6.03E-04	N/A	N/A
Cl-36	2.50E-04	2.50E-04	9.50E-01	2.64E-04
Cm-243	1.29E-04	9.49E-05	2.60E-02	3.65E-03
Cm-244	8.80E-03	5.35E-03	9.40E+01	5.69E-05
Cm-245	2.05E-03	2.05E-03	5.60E-01	3.66E-03
Cm-247	8.39E-04	8.39E-04	N/A	N/A
Cm-248	2.70E-14	2.70E-14	N/A	N/A
Co-60	3.92E-04	7.07E-05	7.10E+00	9.96E-06
Cs-135	7.58E-03	7.58E-03	9.00E-03	8.42E-01
Cs-137	9.42E+02	6.98E+02	2.30E+05	3.04E-03
Eu-152	5.33E-05	2.74E-05	7.80E+00	3.51E-06
Eu-154	1.46E-03	5.13E-04	9.60E+01	5.34E-06
H-3	1.50E+00	7.24E-01	8.20E+02	8.83E-04
I-129	4.36E-02	4.36E-02	9.60E-02	4.54E-01
K-40	2.50E-04	2.50E-04	9.50E-01	2.64E-04
Nb-93m	1.21E-01	9.43E-02	N/A	N/A
Nb-94	4.45E-04	4.45E-04	5.40E-04	8.24E-01
Ni-59	3.04E-04	3.04E-04	2.60E+00	1.17E-04
Ni-63	1.50E-02	1.37E-02	1.80E+02	7.63E-05
Np-237	1.42E-02	1.42E-02	2.70E+01	5.25E-04
Pa-231	3.28E-07	3.99E-07	1.50E-01	2.66E-06
Pd-107	1.36E-03	1.36E-03	9.50E-01	1.43E-03
Pt-193	3.67E-01	3.07E-01	7.30E+01	4.20E-03
Pu-238	3.16E+01	2.85E+01	1.50E+03	1.90E-02
Pu-239	7.86E-01	7.86E-01	1.50E+03	5.24E-04
Pu-240	7.86E-01	7.85E-01	1.50E+03	5.23E-04
Pu-241	1.01E+01	5.38E+00	1.60E+04	3.36E-04
Pu-242	7.69E-02	7.69E-02	4.50E+00	1.71E-02
Pu-244	3.57E-04	3.57E-04	2.10E-02	1.70E-02
Ra-226	4.84E-08	2.31E-07	1.30E+01	1.78E-08
Ra-228	2.21E-06	2.21E-06	2.30E-02	9.62E-05
Se-79	4.18E-02	4.18E-02	8.90E+01	4.69E-04
Sm-151	3.95E-02	3.58E-02	3.70E+03	9.67E-06

Table 4.1-10: Saltstone Disposal Facility SDU Cell 3A Inventory Comparison (Continued)

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [d] (Ci)	Evaluated Total Inventory per SDU[b] 10/1/2032 (Ci)	Comparison Ratio
Sn-126	5.67E-01	5.67E-01	3.90E+02	1.45E-03
Sr-90	3.80E+01	2.78E+01	3.60E+06	7.72E-06
Tc-99	5.70E+01	5.70E+01	5.40E+02	1.05E-01
Th-229	1.35E-04	3.56E-04	1.70E+01	2.09E-05
Th-230	2.51E-05	4.02E-05	1.30E+01	3.09E-06
Th-232	2.21E-06	2.21E-06	2.30E-01	9.62E-06
U-232	3.37E-03	2.96E-03	6.80E-02	4.35E-02
U-233	1.95E-01	1.95E-01	1.60E+01	1.22E-02
U-234	1.26E-01	1.27E-01	9.30E+00	1.37E-02
U-235	2.60E-04	2.60E-04	3.80E-01	6.84E-04
U-236	1.39E-03	1.39E-03	4.30E-01	3.23E-03
U-238	4.75E-03	4.75E-03	1.60E+01	2.97E-04
Y-90[c]	3.80E+01	2.78E+01	N/A	N/A
Zr-93	6.08E-02	6.08E-02	9.50E+01	6.39E-04

N/A – Radionuclide not evaluated in the FY2014 SDF SA (SRR-CWDA-2014-00006) per SRR-CWDA-2013-00147

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Evaluated total inventories from FY2014 SDF SA (SRR-CWDA-2014-00006).

[c] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[d] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

Table 4.1-11: Saltstone Disposal Facility SDU Cell 6 Inventory Comparison

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [d] (Ci)	Evaluated Total Inventory per SDU[b] 10/1/2032 (Ci)	Comparison Ratio
Ac-227	4.21E-07	5.65E-07	8.40E-02	6.72E-06
Al-26	4.80E-04	4.80E-04	2.40E+01	2.00E-05
Am-241	3.76E-02	3.95E-01	1.40E+04	2.82E-05
Am-242m	7.43E-05	6.97E-05	9.10E+00	7.66E-06
Am-243	7.38E-04	7.37E-04	6.90E-02	1.07E-02
Ba-137m[c]	1.99E+03	1.47E+03	N/A	N/A
C-14	1.29E+00	1.29E+00	5.80E+02	2.23E-03
Cf-249	3.72E-13	3.63E-13	N/A	N/A
Cf-251	1.27E-14	1.26E-14	N/A	N/A
Cl-36	8.09E-04	8.09E-04	9.50E-01	8.51E-04
Cm-243	3.30E-04	2.42E-04	2.60E-02	9.31E-03
Cm-244	3.60E-02	2.19E-02	9.40E+01	2.33E-04
Cm-245	1.93E-03	1.93E-03	5.60E-01	3.45E-03
Cm-247	6.43E-14	6.43E-14	N/A	N/A
Cm-248	6.71E-14	6.71E-14	N/A	N/A
Co-60	4.81E-04	8.67E-05	7.10E+00	1.22E-05
Cs-135	1.15E-02	1.15E-02	1.40E-01	8.22E-02
Cs-137	2.10E+03	1.56E+03	2.30E+05	6.78E-03
Eu-152	1.32E-04	6.80E-05	7.80E+00	8.72E-06
Eu-154	1.51E-03	5.27E-04	9.60E+01	5.49E-06
H-3	3.50E+00	1.69E+00	8.20E+02	2.06E-03
I-129	8.96E-02	8.96E-02	1.60E+00	5.60E-02
K-40	2.27E-03	2.27E-03	9.50E-01	2.39E-03
Nb-93m	3.08E-01	2.80E-01	N/A	N/A
Nb-94	8.52E-04	8.52E-04	5.40E-04	1.58E+00
Ni-59	6.51E-04	6.51E-04	2.60E+00	2.50E-04
Ni-63	3.24E-02	2.96E-02	1.80E+02	1.65E-04
Np-237	4.00E-02	4.00E-02	2.70E+01	1.48E-03
Pa-231	7.58E-07	9.23E-07	1.50E-01	6.16E-06
Pd-107	3.38E-03	3.38E-03	9.50E-01	3.56E-03
Pt-193	9.26E-01	7.73E-01	7.30E+01	1.06E-02
Pu-238	7.47E+01	6.74E+01	1.90E+04	3.55E-03
Pu-239	1.92E+00	1.92E+00	9.60E+03	2.00E-04
Pu-240	1.92E+00	1.92E+00	1.90E+03	1.01E-03
Pu-241	2.35E+01	1.25E+01	1.60E+04	7.84E-04
Pu-242	2.17E-01	2.17E-01	4.50E+00	4.82E-02
Pu-244	1.01E-03	1.01E-03	2.10E-02	4.80E-02
Ra-226	1.25E-07	1.24E-06	1.30E+01	9.51E-08
Ra-228	8.45E-06	8.44E-06	2.30E-02	3.67E-04
Se-79	7.76E-02	7.76E-02	8.90E+01	8.71E-04
Sm-151	1.11E-01	1.00E-01	3.70E+03	2.71E-05

Table 4.1-11: Saltstone Disposal Facility SDU 6 Inventory Comparison (Continued)

Radionuclide	Current Inventory [a] 9/30/2019 (Ci)	Current Inventory Decayed to 10/1/2032 [d] (Ci)	Evaluated Total Inventory per SDU[b] 10/1/2032 (Ci)	Comparison Ratio
Sn-126	1.34E+00	1.34E+00	3.90E+02	3.43E-03
Sr-90	1.23E+02	8.97E+01	3.60E+06	2.49E-05
Tc-99	1.35E+02	1.35E+02	3.70E+03	3.65E-02
Th-229	2.25E-03	2.87E-03	1.70E+01	1.69E-04
Th-230	1.76E-04	2.20E-04	1.30E+01	1.69E-05
Th-232	8.45E-06	8.45E-06	2.30E-01	3.67E-05
U-232	6.43E-03	5.64E-03	6.80E-02	8.29E-02
U-233	5.50E-01	5.49E-01	1.60E+01	3.43E-02
U-234	3.62E-01	3.65E-01	9.30E+00	3.92E-02
U-235	6.01E-04	6.01E-04	3.80E-01	1.58E-03
U-236	3.83E-03	3.83E-03	4.30E-01	8.92E-03
U-238	1.00E-02	1.00E-02	1.60E+01	6.28E-04
Y-90[c]	1.23E+02	8.98E+01	N/A	N/A
Zr-93	2.49E-01	2.49E-01	9.50E+01	2.62E-03

N/A – Radionuclide not evaluated in the FY2014 SDF SA (SRR-CWDA-2014-00006) per SRR-CWDA-2013-00147

[a] Total inventories which include decay and ingrowth from SRR-CWDA-2019-00110.

[b] Evaluated total inventories from FY2014 SDF SA (SRR-CWDA-2014-00006).

[c] Data included for inventory only, radionuclide not included in SDF modeling due to short half-life (SRNL-STI-2009-00115 Section 4.2).

[d] Decayed inventory extracted from SDF-WIDE model, Version 1.11.

5. MONITORING

The environmental monitoring and disposal unit inspection programs were developed to be consistent with the 2009 SDF PA. The monitoring data evaluation is presented in this section.

5.1 Reason for Monitoring

Per the requirements in the DAS issued for the SDF (WDPD-12-49), a monitoring plan shall be written, approved, and implemented within one year of issuance of the DAS and updated at least every five years. This monitoring plan includes annual data review and evaluation. Following this annual data review and evaluation, any modifications to this monitoring plan that may be applicable will be noted and the plan updated as necessary. *Performance Assessment Monitoring Plan for the Saltstone Disposal Facility at the Savannah River Site*, SRR-CWDA-2013-00026, Rev. 1 satisfies this monitoring plan requirement.

Monitoring to be performed as part of this plan is intended to meet the requirements of DOE O 435.1, Chg. 1 and its associated implementation Manual and Guide. These documents require disposal facilities to monitor for compliance with the conditions of the DAS. In particular, the following must be addressed:

- The site-specific 2009 SDF PA (SRR-CWDA-2009-00017) and associated composite analysis (CA) (SRNL-STI-2009-00512) were used to determine the media, locations, radionuclides, and other substances to be monitored.
- The environmental monitoring program includes measuring and evaluating releases, migration of radionuclides, SDU subsidence, and changes in disposal facility and disposal site parameters, which may affect long-term performance.
- The environmental monitoring program is capable of detecting changing trends in performance to allow application of any necessary corrective action prior to exceeding the PA performance objectives (DOE M 435.1-1).

5.2 Monitoring Plan

Table 5.2-1 summarizes the monitoring implemented to assess the SDF compliance with the pertinent performance objectives as presented in the *Performance Assessment Monitoring Plan for the Saltstone Disposal Facility at the Savannah River Site*, SRR-CWDA-2013-00026, Rev. 1. Figure 5.2-1 shows the monitoring locations.

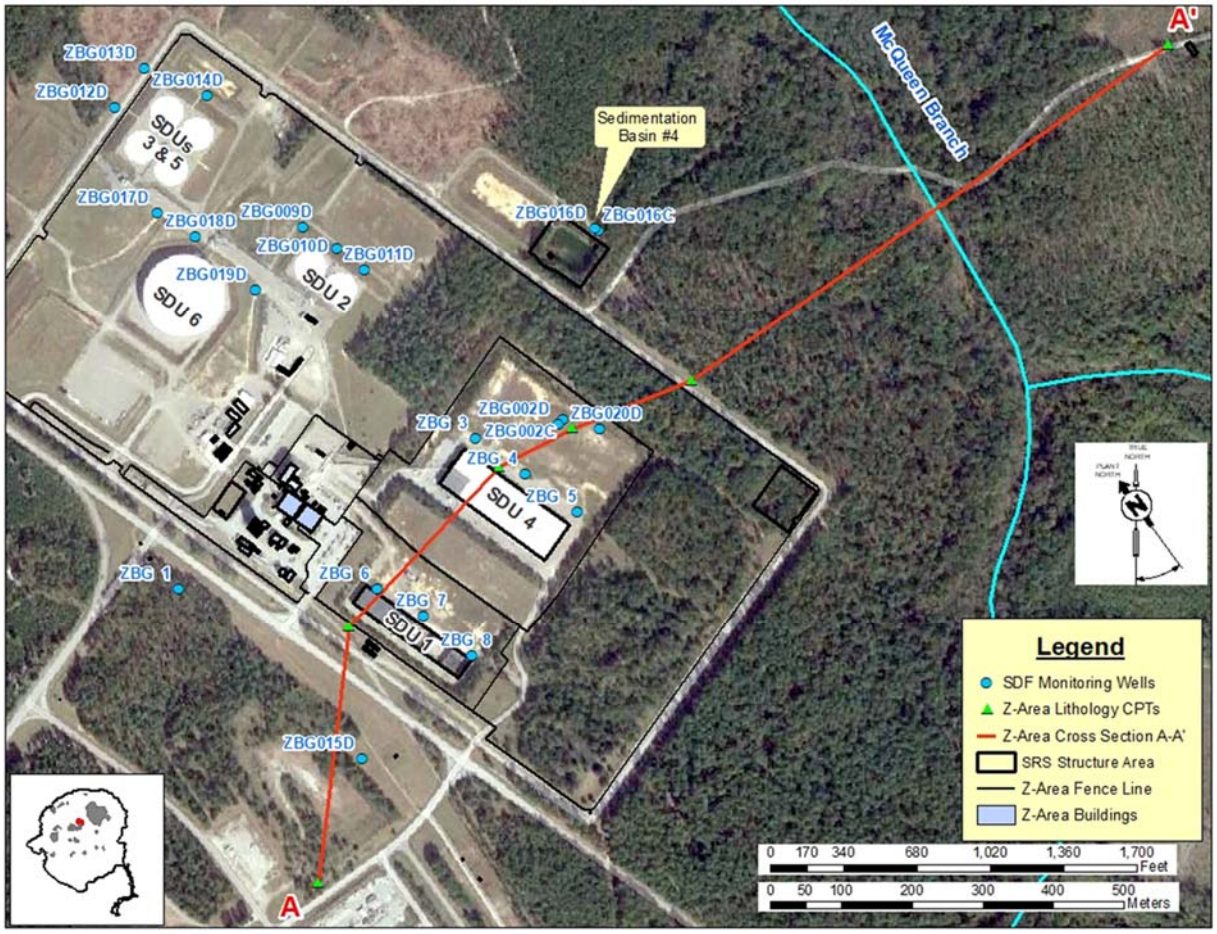
In addition to exposure pathways, certain facility features are relevant to monitoring the release of constituents to the surrounding environment. Per the existing SDF Monitoring Plan, quarterly visual inspection of the SDU integrity is sufficient to indicate conditions that may affect SDU integrity. [SRR-CWDA-2013-00026] The SDF has met this requirement by creating an *Inspection Program Plan for Z-Area Vault 4* (LWO-LWE-2008-00023) for SDU 4 due to its existing condition. In-service inspection of SDU 4 will be performed to provide a historical photographic record of the external SDU condition. Inspections will be made of SDU 4 to document SDU cell wall coating and wall conditions, including baseline inspections. SDUs 2, 3, and 5 are buried such that inspections of the roof are performed during routine operator rounds. The roof and walls of SDU 6 are inspected during routine operator rounds.

Table 5.2-1: Summary Monitoring Table

Pathway/ Relevant Feature	Media Features/ Inspection	Monitoring Location	Radionuclide/Other Substance	Sampling Frequency	Sampling Method	Analytical Method	Minimum Detectable Activity/Method Detection Limit
Waste Acceptance Criteria (WAC) Transfer Compliance	Tank 50 content in compliance with Saltstone WAC	Completed evaluation on file in the Control Room	Tank 50 content as provided in the Waste Characterization System prior to transfer to SPF	WAC Evaluation complete prior to transfer from Tank 50 to SPF	N/A	N/A	Tank 50 content in compliance with Saltstone WAC
SDU integrity	Visual and video inspection of SDUs	Periphery and interior, if accessible, of SDUs	N/A	Monthly, at a minimum	Visual inspections, Video camera	N/A	N/A
Water resource protection	Groundwater	Well clusters ZBG 1 through ZBG 20	Nitrate (nitrate/nitrite) Gross alpha Nonvolatile beta Beta/photon emitters I-129 Tc-99 Tritium	Twice per year	Well sampling	As designated in the groundwater monitoring plan (WSRC-TR-2005-00257)	As designated in the groundwater monitoring plan (WSRC-TR-2005-00257)
		Well clusters ZBG 1 through ZBG 20	Radium-226 Radium-228 Benzene Toluene Tetrachloroethylene Trichloroethylene	Once every two years (odd Fiscal years, e.g., FY2019)	Well sampling	As designated in the groundwater monitoring plan (WSRC-TR-2005-00257)	As designated in the groundwater monitoring plan (WSRC-TR-2005-00257)

N/A - Not Applicable
[SRR-CWDA-2013-00026]

Figure 5.2-1: SDF Existing Monitoring Well Locations



5.3 Evaluation of Monitoring Data

Contamination resulting from previous operational upsets has resulted in minor groundwater contamination being detected below SDU 4. In addition, in 2014, due to concerns of potential groundwater contamination from Sedimentation Basin No. 4, a shallow well (ZBG016D) and a deeper well (ZBG016C) were installed to monitor perched water in the vadose zone and the groundwater. In 2017, four additional wells (ZBG017D, ZBG018D, ZBG019D, and ZBG020D) were added to the SDF monitoring network. Wells ZBG017D, ZBG018D, and ZBG019D were added to monitor SDU 6, and in the future, SDUs 7 and 8. These wells helped establish background chemistry as SDU 6 was just placed into service in August 2018. Well ZBG020D provides additional downgradient groundwater monitoring of SDU 4. The SDF groundwater monitoring well network and monitoring plan, *Groundwater Monitoring Plan for the Z-Area Saltstone Disposal Facility,* WSRC-TR-2005-00257, are designed to effectively detect any release associated with SDUs in the SDF.

Per, the *Z-Area Saltstone Disposal Facility Groundwater Monitoring Report for 2019* (SRNS-TR-2019-00326), groundwater samples were collected during 1Q Calendar Year (CY) 2019 and 3QCY19 from 21 of the 22 wells monitoring the SDF (monitoring well ZBG016D was dry). The

Z-Area groundwater well samples were sent to South Carolina Department of Health and Environmental Control (SCDHEC) certified lab for analyses. Groundwater monitoring results were compared to Practical Quantitation Limits (PQLs), background concentrations, and Groundwater Protection Standards (GWPS). PQLs are indicators of laboratory instrument sensitivity, but are not regulatory limits, nor are they risk-based. The PQL is the lowest concentration of an analyte which can be reliably quantified in a given sample. Background concentrations are based on historical data from two wells (ZBG 1 and ZBG015D) upgradient of the SDF. Data from ZBG015D is used for background comparisons to monitoring data collected at wells downgradient of SDUs 1 and 4. Data from ZBG 1 is used for background comparisons to monitoring data collected at wells downgradient of SDUs 2A, 2B, 3A, 3B, 5A, 5B, and 6. Wells ZBG017D, ZBG018D, ZBG019D, and ZBG020D were sampled after installation in the second quarter of 2017. The second quarter 2017 samples from ZBG017D through ZBG019D helped establish background groundwater chemistry for SDUs 6, 7, and 8.

In 2019, three constituents (nonvolatile beta (NVB), Tc-99, and nitrates) continued to have detected results in four monitoring wells: ZBG002C, ZBG002D, ZBG 4, and ZBG020D. Results for the sampling from FY2019 are summarized in Table 5.3-1. As conductivity is also an indication of potential contamination, its results are also included in Table 5.3-1.

Table 5.3-1: SDF Monitoring Wells with Detected Concentrations in 2019

Saltstone Monitoring Well	NVB (pCi/L)		Tc-99 (pCi/L)		Conductivity (uS/cm)		Nitrate (mg/L)	
	1QCY19	3QCY19	1QCY19	3QCY19	1QCY19	3QCY19	1QCY19	3QCY19
ZBG002C	43.4	41.9	62.9	99.5	49	45	3.25	3.21
ZBG002D	74.2	18.1	87.1	36.8	NR	39	5.44	2.32
ZBG 4	8.65	4.38	3.89	3.41	26	24	0.96	1.03
ZBG020D	48.5	16.0	94.9	27.4	31	26	1.26	2.02

NR – Not Reported

Water Table Wells Downgradient of SDU 4

In 3QCY19, groundwater at well ZBG002D showed decreases in nonvolatile beta activity (18.1 pCi/L), Tc-99 activity (36.8 pCi/L), nitrate concentration (2.32 mg/L), and conductivity (39 µS/cm) relative to the 3QCY18 concentrations for nonvolatile beta activity (100 pCi/L), Tc-99 activity (164 pCi/L), nitrate concentration (7.42 mg/L), and conductivity (69.0 µS/cm). The 3QCY19 results for ZBG002D are lower than the maximum ZBG 2 concentrations in 2015 for nonvolatile beta activity (158 pCi/L), Tc-99 activity (238 pCi/L), nitrate concentration (9.9 mg/L), and conductivity (102 µS/cm).

In 2019, the ZBG020D well samples had slightly lower concentrations of nonvolatile beta activity (16 pCi/L), Tc-99 activity (27.4 pCi/L), nitrate concentration (2.02 mg/L), and conductivity (26 µS/cm) compared to the 2019 ZBG002D samples. These data indicate the highest concentration portion of the groundwater plume is located around wells ZBG002D and ZBG020D. In 2019, Tc-99 and nitrate groundwater concentrations at wells ZBG002D and ZBG020D remained below their respective GWPS. [SRNS-TR-2019-00326]

Nonvolatile Beta Speciation

The 1QCY19 and 3QCY19 nonvolatile beta activities of wells ZBG002D, ZBG002C, and ZBG020D and 1QCY19 results for ZBG 4 exceeded the 8 pCi/L action level (presented as a GWPS), as shown in Table 5.3-2. As stated in the *Groundwater Monitoring Plan for the Z-Area Saltstone Disposal Facility,* WSRC-TR-2005-00257, if the nonvolatile beta result exceeds 8 pCi/L at a given well, the well will be resampled within 30 days and a Sr-90 analysis will be run. If Sr-90 is detected, it will be added to the monitoring list in Table 2 of the groundwater monitoring plan.

If the nonvolatile beta result exceeds 30 pCi/L (the standard for Ru- 106) at a given well, then that well and the associated background well (ZBG 1 or ZBG 15) will be resampled within 30 days and analyses for nonvolatile beta and for all constituents in Table 3 of the groundwater monitoring plan will be run.

Any nuclides detected above the associated maximum background concentration of the GWPS will be added to the monitoring list in Table 2 of the groundwater monitoring plan. If the follow-up sampling exceeds the maximum background concentration and GWPS, a plan for assessing the lateral and vertical extent of the plume will be developed and submitted to SCDHEC within 60 days. [WSRC-TR-2005-00257]

Sr-90 sample results for ZBG002C, ZBG002D, ZBG020D, and ZBG 4 were below their minimum detection limits indicating Sr-90 was not detected in these samples. [SRNS-TR-2019-00326] The 1QCY19 and 3QCY19 nonvolatile beta results for ZBG002D, ZBG002C, and ZBG020D also exceeded the 30 pCi/L threshold for nonvolatile beta, so the Contingent Analyses 2 constituents from SRNS-TR-2019-00326, Table 1, were required for these wells as well as the background well ZBG015D samples. Only Tc-99 was detected in the ZBG002C, ZBG002D, and ZBG020D contingent samples from the additional analyses which has previously been identified as a contaminant to monitor. [SRNS-TR-2019-00326]

Table 5.3-2: SDF PA Nonvolatile Beta Compliance Monitoring

Saltstone Monitoring Well	Monitoring Type	Monitoring Results	Performance Objective Measure or Other Regulatory Limit	Action Level	Action Taken	PA Impact
ZBG002C	Groundwater Nonvolatile Beta	43.4 pCi/L	8 pCi/L	30 pCi/L	Sr-90 Analyses; Tc-99 Analysis	None
ZBG002D	Groundwater Nonvolatile Beta	74.2 pCi/L	8 pCi/L	30 pCi/L	Sr-90 Analyses; Tc-99 Analysis	None
ZBG 4	Groundwater Nonvolatile Beta	8.65 pCi/L	8 pCi/L	30 pCi/L	Sr-90 Analyses	None
ZBG020D	Groundwater Nonvolatile Beta	48.5 pCi/L	8 pCi/L	30 pCi/L	Sr-90 Analyses Tc-99 Analysis	None

In 2014, well ZBG002C was installed adjacent to ZBG002D with a screen zone below the Tan Clay Confining Zone (TCCZ) in the Upper Three Runs Aquifer – Lower Aquifer Zone (UTRA-LAZ) to monitor groundwater in the LAZ. In 2019, samples collected in the LAZ at wells ZBG002C, ZBG 3, ZBG 4, and ZBG016C had levels of nonvolatile beta groundwater concentrations (3.14-43.4 pCi/L) that exceeded the historic maximum nonvolatile beta value (2.17 pCi/L) for background well ZBG015D. The nonvolatile beta data, in conjunction with the nitrate and Tc-99 data, for wells ZBG002C, ZBG 3 and ZBG 4 indicate contamination below the TCCZ. In contrast, the nonvolatile beta activity (3.14 pCi/L) in the groundwater at ZBG016C in 1Q19 was not accompanied by other contaminants. Cs-137, Tc-99, I-129, and nitrates were below detection limits for the 1Q19 sample from the ZBG016C, but naturally-occurring Pb-214 and Bi-214 were above their respective PQLs for that sample. The nonvolatile beta activity in the groundwater at ZBG016C appears to be naturally-occurring, and not from contaminants in Sedimentation Basin No. 4. The SDF groundwater monitoring well network is adequately monitoring contaminants above and below the TCCZ per the *Z-Area Saltstone Disposal Facility Groundwater Monitoring Report for 2019* (SRNS-TR-2019-00326).

At well ZBG 6 the 3Q19 nonvolatile beta result (3.7 pCi/L) exceeds the historic maximum nonvolatile beta value (2.17 pCi/L) for background well ZBG015D. The 3Q19 nonvolatile beta result is a decrease relative to the 1Q19 sample result (4.37 pCi/L) at ZBG 6. Conductivity and nitrate concentrations have also slowly increased in the groundwater at this well over the past 2 years, but so far Tc-99 has not been detected in any of the ZBG 6 samples.

In 2019, ZBG 6 (1Q19) and ZBG020D (3Q19) had the highest sum of beta-emitting radionuclides greater than their respective PQLs, but the 1Q19 and 3Q19 sum of beta-emitting activity levels (1.49 mrem and 0.48 mrem, respectively) for ZBG 6 and ZBG020D are below the GWPS (4 mrem). The beta-emitting radionuclides, including all the contingent analyses constituents, above their respective PQLs were Bi-214, Pb-214, Tc-99 and tritium.

Z-Area Sedimentation Basin No. 4

Two groundwater monitoring wells (ZBG016C and ZBG016D) have been installed downgradient of Sedimentation Basin No. 4, one to monitor perched water above the TCCZ (ZBG016D) and one to monitor the water table below the TCCZ (ZBG016C). The 1QCY19 and 3QCY19 data for ZBG016C indicates there not have been any impacts to the groundwater at Sedimentation Basin No. 4. The screen zone for well ZBG016D has been dry since installation, indicating there was no perched water at this location in 1QCY19 and 3QCY19 (SRNS-TR-2019-00326). Analytical results from the groundwater sampling from 1QCY19 and 3QCY19 are presented in the *Z-Area Saltstone Disposal Facility Groundwater Monitoring Report for 2019* (SRNS-TR-2019-00326). The groundwater analytical results to date do not contradict 2009 SDF PA model estimates.

Tc-99

In 2019, the maximum Tc-99 groundwater concentration (99.5 pCi/mL) was in 3Q19 at well ZBG002C. The Tc-99 groundwater concentrations did not exceed the GWPS (900 pCi/L) at any of the ZBG wells. Wells ZBG002C, ZBG002D and ZBG020D indicate the location of the highest concentration of the Tc-99 groundwater plume. The changing concentrations of Tc-99, nonvolatile beta, nitrate and specific conductance in the Upper Aquifer Zone (UAZ) may, in part, be due to increased and decreased rainwater infiltration, as these constituents tend to inversely correlate with water elevation measurements at wells ZBG 2 and ZBG002D.

The samples from well ZBG002C had Tc-99 results above the detection limit and PQL in 1Q19 (62.9 pCi/L) and 3Q19 (99.5 pCi/L), which is consistent with the nonvolatile beta results from the same well samples and historical data. The 2014 through 2019 Tc-99 data from wells ZBG002C and ZBG 4 indicate Tc-99 contamination has migrated through the TCCZ (SRNS-TR-2019-00326).

Nitrates

In 2019, the nitrate sample from well ZBG002D (1Q19) had the highest nitrate groundwater concentration (5.44 milligrams/liter [mg/L]), while the sample from well ZBG017D (1Q19) had the second highest groundwater concentration (5.05 mg/L) for nitrates. The ZBG017D and the ZBG002D nitrate sample results did not exceed the GWPS (10 mg/L). However, the results for ZBG017D and ZBG002D samples did exceed the PQLs and maximum concentrations from background wells ZBG 1 and ZBG015D (2.03 mg/L in 2006 and 1.30 mg/L in 2015, respectively). The 1Q19 nitrate groundwater concentrations at well ZBG002D increased (5.44 mg/L) but decreased in 3Q19 (2.32 mg/L) relative to the 2018 nitrate groundwater concentrations at ZBG002D. Conductivity and nonvolatile beta groundwater concentrations showed similar trends at well ZBG002D in 2019 (SRNS-TR-2019-00326).

The elevated nitrate groundwater concentration at well ZBG017D may be from an upgradient source because SDU 6 was placed in service in August 2018, and the upgradient well ZBG 1 has a history of elevated nitrate concentrations. However, the maximum nitrate value at background

well ZBG 1 was 2.03 mg/L in 2006. Alternatively, the elevated groundwater nitrate concentrations at well ZBG017D could be, in part, from the release of the National Science Foundation approved dye (Rhodamine WT) after the leak tests at SDU 6, as the dye contains nitrogen. Wells ZBG019D and ZBG009D, which are near ZBG017D, also have elevated nitrate concentrations. The 1Q19 samples from ZBG009D, ZBG017D and ZBG0019D did not have detectable levels of Tc-99, and the nonvolatile beta results are below PQLs, which indicates the source of the nitrates is not from saltstone material (SRNS-TR-2019-00326).

In 1Q19, nitrate groundwater concentrations increased to 1.98 mg/L at well ZBG 6, which is greater than the PQL and maximum nitrate background value (1.30 mg/L) at ZBG015D, but less than the GWPS (10 mg/L). The groundwater conductivity measurements at well ZBG 6 have also been gradually increasing since 2015. The nonvolatile beta groundwater concentrations were above the PQL in 3Q19 (3.7 pCi/L), though Tc-99 remains below the detection limit at ZBG 6. The nitrate, conductivity and nonvolatile beta results at ZBG 6 may be early indicators of contamination from historic spills (SRNS-TR-2019-00326).

Tritium

Low concentrations of tritium are present in nearly all SDF monitoring wells, including the two background wells and does not appear to be related to activities associated with the SDF. [SRNS-TR-2019-00326] The maximum tritium concentrations were 4.27 pCi/mL and 3.03 pCi/L at monitoring well ZBG016C in 1Q19 and 3Q19, respectively. The 1Q19 concentration is above the historic maximum (4.02 pCi/mL) for background well ZBG015D, but below the GWPS (20 pCi/mL). The older background well ZBG 1 has indicated steadily decreasing tritium concentration trends from 19.0 pCi/mL in 1987 to 1.21 pCi/mL in both 1Q19 and 3Q19 (SRNS-TR-2019-00326).

In summary, groundwater concentrations in the UAZ downgradient of SDU 4 continued to decrease from 3Q15, when the highest historical concentrations were measured. Downgradient of SDU 4, contaminants (specifically, Tc-99, nonvolatile beta, and nitrates) have begun to move downward into the LAZ, but concentrations remain below their respective GWPS. There also have been no impacts to groundwater from Sedimentation Basin No. 4. Overall, no new contamination was identified at the SDF during 2019 and no constituents monitored exceed their respective GWPSs, with the exception of the nonvolatile beta action levels. Additional sampling, required by the action level exceedances, resulted in no constituents exceeding their respective GWPS.

6. RESEARCH AND DEVELOPMENT

Several studies from FY2019 have continued research on properties considered critical to the performance of saltstone and are described in the sections below and summarized in Table 6-1. A more in-depth discussion of on-going and future studies can be found in the *Savannah River Site Liquid Waste Facilities Performance Assessment Maintenance Program FY2019 Implementation Plan*. [SRR-CWDA-2018-00092]

Table 6-1: Summary of FY2019 R&D Activities

R&D Document Number	Results/Discussion	PA/CA Impacts
SRRA099188-000010	**Radionuclide Leaching Characteristics from Saltstone Monolith** Dynamic Leaching Method (DLM) testing continued for one of the SDU Cell 2A samples (experiment began in March 2016). Initially the concentration of Tc-99 in the SDU 2A core leachate was fairly consistent around a value of 2E-07 mol/L but dropped to around 5E-08 mol/L once the pH of the leachate dropped below 11 and has remained there since. The consistency of the Tc-99 concentrations measured in the leachate, both before and after the pH drops below 11, supports the current modeling approach that Tc-99 release from saltstone is initially solubility controlled. EPA Method 1315 and DLM experiments were conducted in FY2019 on four new saltstone simulants spiked with both Tc-99 and I-129 and synthesized using Lehigh Ground Granulated Blast Furnace Slag (GGBFS). Two of the saltstone simulants were prepared using the historical saltstone formulation (i.e., 45 wt% GGBFS, 45 wt% Class F Fly Ash (FA), and 10 wt% ordinary portland cement (OPC)) while the other two were prepared using the cement-free formulation (i.e., 60 wt% GGBFS and 40 wt% FA) proposed in the *Cement-Free Formulation Down-Select Report*. [SRR-CWDA-2019-00003] Results from these experiments, as well as reduction capacity testing performed on non-spiked samples, suggest that the cement-free saltstone formulation is comparable in performance, from a PA perspective (i.e., leaching and saturated hydraulic conductivity), to the historically used 45 wt% GGBFS, 45 wt% FA, and 10 wt% OPC saltstone formulation. Comparison of the Tc-99 DLM effluent data collected from the historical and cement-free saltstone samples as well as the SDU 2A core with theoretical solubility curves for Tc solid phases suggests that under reducing conditions, Tc release from saltstone is solubility-controlled by one or more Tc(IV)-oxide solid phases ($TcO_2 \cdot x(H_2O)$).	To be determined by FY2020 PA activities.

Table 6-1: Summary of FY2019 R&D Activities (Continued)

R&D Document Number	Results/Discussion	PA/CA Impacts
In preparation	**Long-Term Radiological Lysimeter Program** This ongoing lysimeter effluent testing, in conjunction with solid phase analysis of the lysimeter cores and source material, provides support modeling of transport, including K_d values. In FY2019, eleven new lysimeters, containing plutonium, neptunium, or radium sources were installed at the Radionuclide Field Lysimeter Experiment (RadFLEx) facility at SRS.	

6.1 Radionuclide Leaching Characteristics from Saltstone Monolith

Experiments of radionuclide leaching have been ongoing since FY2015. The purpose of this study is two-fold. The first element of the investigation was to evaluate test methods with respect to saltstone leaching utilizing rhenium (Re) and stable iodine (I-127) as non-radioactive surrogates for Tc-99 and I-129. Test methods employed included a standardized semi-dynamic leaching test, U.S. Environmental Protection Agency (EPA) Method 1315, *Mass Transfer Rates of Constituents in Monolithic or Compacted Granular Materials Using a Semi-Dynamic Tank Leaching Procedure*, and a dynamic leaching test was developed as part of this scope. [EPA_Method_1315] The dynamic leaching methodology uses the flexible-wall permeameter apparatus that was more commonly used for measuring the Saturated Hydraulic Conductivity (SHC) of saltstone. The intent is to force leachate through the interior of the saltstone monolith to mimic the eventual ingress of water into saltstone and subsequent pore volume exchange to establish the dynamic leaching behavior of saltstone contaminants. The second element of the investigation was to utilize the tests methods (optimized with non-radioactive surrogates) for characterizing the leaching behavior of saltstone samples spiked with Tc-99 and I-129 in addition to the characterization of saltstone cores retrieved from SDU Cell 2A.

This task will provide empirical leaching (diffusion) data for Tc-99, I-129, and potentially other saltstone contaminants that can be used as direct inputs to the saltstone transport models. In addition, the development of a dynamic leaching test will provide new information regarding the leaching of saltstone associated with multiple pore volume exchanges.

In FY2015, the standardized semi-dynamic leaching, EPA Method 1315, was used to evaluate the leaching characteristics of Technetium (Tc)-spiked saltstone samples cured for three and six months. [EPA_Method_1315] A Tc-spiking concentration of 1.2E-02 mmol/L was utilized for all samples, which is consistent with Tc levels detected in Tank 50 waste solution. Tests were conducted with an artificial groundwater solution equilibrated with respect to three different atmospheres (oxic, anoxic, and reducing). The leaching rate of Tc-99 was observed to decrease over the course of testing but exhibited no clear response to either curing duration (3 or 6 months) or test atmosphere (oxic, anoxic, and reducing). In comparison to Re-spiked samples, the proportion of Tc leached during testing was significantly less than Re (i.e., <0.75 % for Tc compared to 6-9 % for Re). Rhenium is typically considered as a non-radioactive surrogate for Tc but this data and known differences in redox potential between the two elements, imply that Re does not exhibit equivalent leaching behavior to Tc and as such, it may not be a legitimate surrogate. The effective diffusivities determined via EPA testing for Tc and Re were 2.0-3.7E-10

cm^2/sec and 2.7-3.5E-08 cm^2/sec, respectively. A value of 1.0E-08 cm^2/sec was used in the FY2014 SDF SA, which is conservative.

Dynamic leaching tests utilizing the method developed using Re-spiked samples in FY2014 were conducted on Tc-spiked samples in FY2015. Based on the occurrence of effluent degassing in FY2014, which appeared to prevent the acquisition of consistent data, a bladder accumulator was incorporated into the experimental set up to restrict contact between the "test" solution saturating the column and the solution in the permeameter. This modification was successful in excluding dissolved gases from the pressurized inlet solution, and subsequent bubble formation in the sample and in the effluent solution. Similar to the EPA measurements, technetium and rhenium were spiked at levels equivalent to those measured in Tank 50 waste. After collecting approximately 45 to 50 mL of leachate for each of the Tc- and Re-spiked samples, the cumulative proportions of technetium and rhenium (as a percentage of the original spike amounts) leached were 0.3 % and 10 %, respectively. The disparity in the data again suggests that rhenium may not be a suitable non-radioactive surrogate for technetium at least in regards to leaching behavior. Based on sample size and a measured porosity of 60 %, 45 to 50 mL equates to the exchange of approximately 1 to 2 pore volumes, though it is acknowledged that not all of the porosity may be involved in liquid transport through the sample. [SREL Doc. R-15-0003]

Both EPA Method 1315 and dynamic leaching testing were continued in FY2017 and encompassed evaluation of radionuclide-spiked saltstone simulants and actual saltstone cores extracted from SDU Cell 2A. The data from these studies is provided in SREL Doc. R-16-0003 and SREL Doc. R-17-0005.

With respect to EPA Method 1315, Tc-99 leaching rates for the spiked saltstone samples appeared to be sensitive to curing duration and the reduction capacity of the blast furnace slag (BFS) used in making the grout. Due to supply cessation of a historically utilized BFS, an alternate, or "new", BFS source was sought and approved for use in processing future saltstone batches at SRS. [SRR-CWDA-2015-00072] Longer curing times and higher reduction capacity for the as-received BFS resulted in lower effective diffusivities. A Tc-spiked, simulant saltstone processed using BFS with a reduction capacity of 1,600 μeq/g and cured for 6 months indicated an effective diffusivity of 5.7E-12 cm^2/sec compared to 3.0E-10 cm^2/sec for a sample processed using BFS with a reduction capacity of approximately 700 μeq/g and cured for 3 months. Reduction capacity can vary between different BFS sources due to differences in the concentrations of components, in particular sulfur and iron, which are known reductants. As anticipated leaching rates and effective diffusivities from the simulant samples for poorly sorbing contaminants like NO_3^- were much higher than for Tc-99 (E-07 to E-08 cm^2/sec for nitrate compared to E-10 to E-12 cm^2/sec for Tc-99).

Tc-99 diffusivities (determined via EPA Method 1315 testing) for three intact saltstone samples (retrieved from SDU Cell 2A) ranged from 5.2E-11 cm^2/sec to 6.4E-11 cm^2/sec. These diffusivities are higher than observed for the simulant samples processed using the "new" BFS which is expected since the saltstone emplaced in SDU Cell 2A was produced with the historically-utilized BFS with the lower measured reduction capacity. As with the simulant samples, the effective diffusivities of poorly adsorbed nitrates were approximately two orders of magnitude higher than the Tc-99 diffusivities. I-129 and Cs-137 diffusivities for the three SDU Cell 2A cores ranged 1.0E-08 cm^2/sec to 5.5E-09 cm^2/sec and 1.1E-10 cm^2/sec to 4.9E-10 cm^2/sec, respectively.

The leaching behavior for I-129 was similar to that observed for poorly adsorbed NO_3^- as indicated by the high effective diffusivities for both components. [SREL Doc. R-17-0005]

In FY2018, EPA 1315 leachates collected from two I-129 spiked saltstone simulants were analyzed to obtain effective diffusivities for iodine in saltstone. The two saltstone simulants utilized the same dry feed composition (45% BFS, 45% fly ash (FA), 10% ordinary Portland cement (OPC)), salt solution (Actinide Removal Process (ARP)-Modular Caustic Side Solvent Extraction Unit (MCU) simulant) and curing time (1.5 months) but differed regarding the BFS source used (Lehigh versus Holcim). The resulting effective diffusivities for iodine from this work (1.4E-08 cm^2/sec and 2.8E-08 cm^2/sec, for saltstone simulants synthesized with Lehigh and Holcim BFS, respectively) were comparable to previous EPA 1315 tests conducted in FY2015 on saltstone simulants cured for 3 months and 6 months (2.9E-08 cm^2/sec and 3.0E-08 cm^2/sec, respectively). These effective diffusivities are noticeably higher than those measured from actual SDU 2A cores (2.5E-09 cm^2/sec, 5.5E-09 cm^2/sec, and 1.0E-08 cm^2/sec) which were cured in SDU 2A for approximately 20 months prior to sampling. Interestingly, for each of these studies (i.e., FY2015 simulants, SDU 2A cores, and FY2018 simulants) the effective diffusivity measured for nitrate was comparable to that of iodine, suggesting that iodine is poorly retained in the saltstone matrix. [SRRA099188-000005]

For Dynamic Leaching Method (DLM) testing, the permeameter system was upgraded to accommodate three test samples at the same time by using the laboratory air compressor to provide the driving force for leaching. The three materials tested include a Tc-99-spiked sample described in SREL Doc. R-15-0003, and two actual radioactive saltstone cores extracted from SDU Cell 2A. Measurement of the Tc-99-spiked sample has been ongoing for almost 30 months with approximately 550 mL of leachate being collected from the sample (equivalent to about 15 pore volumes). The concentration of Tc-99 in the leachate from up to approximately 4.5 pore volumes fluctuated between 5E-09 mol/L and 1E-08 mol/L with a cumulative Tc-99 release of 0.6 %. However, leachates collected after the exchange of 4.5 pore volumes indicated a rapid increase in Tc-99 concentration to around 1E-07 mol/L; at the end of FY2016 the cumulative Tc-99 release from the simulant saltstone sample was approaching 2.5 % (after the exchange of 6.25 pore volumes). In contrast, approximately 50 % of the nitrate in the sample has been released which verifies the poor nitrate retention observed for the EPA tests. It is also noteworthy that the nitrate concentration did not indicate a similar spike as compared to Tc-99 after the exchange of approximately 4.5 pore volumes. Phenomena responsible for the sudden increase in Tc-99 leaching have not been established. In addition, the Evaluation Case in the FY2014 SDF SA used 1E-08 mol/L and a sensitivity case was run using 1E-07 mol/L.

During the FY2017 testing of this Tc-spiked saltstone, a process anomaly occurred in which the sample confining pressure was lost and it is possible that the sample was exposed to the atmosphere. The Tc-99 concentration following this anomaly increased by a factor of three (from 1.2E-07 mol/L to 3.6E-07 mol/L). More significant, however, was a concurrent order of magnitude increase in the SHC from 4.0E-10 cm/sec to 4.0E-09 cm/sec. As such there is concern that the sample, and therefore the data, after this event may be compromised. However, given the extended length of time the sample had been tested, a decision was made to continue the analysis. The sample has been evaluated for approximately 27 months. Fifteen pore volumes have been exchanged and the Tc-99 and NO_3^- concentrations are currently 3.3E-08 mol/L and 0.02 mol/L, respectively. The pH of the leachate is around 10.8, and as such the Tc-99 concentration is in good

agreement with the thermodynamically predicted solubility limit of reduced $TcO_2.xH_2O$ phases. With respect to cumulative percent leached from the saltstone sample the Tc-99 is 11% and the NO_3^- is 94%. The SHC did not decrease to its former value and remains around 2.0E-09 cm/sec. [SRRA099188-000005]

The SDU Cell 2A cores indicate significantly disparate properties from each other and from the Tc-spiked simulant sample. With respect to SHC, the Tc-spiked sample has indicated a relatively consistent value that fluctuates ranging between 5E-10 cm/sec and 2E-09 cm/sec. In contrast, one of the SDU Cell 2A cores indicated a high initial SHC of 1E-08 cm/sec which subsequently dropped to <5E-10 cm/sec after 1.5 pore volume exchanges. The second SDU Cell 2A sample started with an SHC of 2E-09 cm/sec which dropped to <1E-10 cm/sec with less than 1 pore volume exchanged. The FY2014 SDF SA modeled an initial SHC of 6.4E-09 cm/sec, which is conservative. Differences in sample SHC pose some potential issues with respect to sample-to-sample comparison. In particular, variations in SHC result in different pore volume exchange rates and therefore different residence times of the leachate within the sample. Varied residence times may subsequently impact contaminant leaching behavior. The evaluation of the SDU Cell 2A samples continued through FY2017 though the sample exhibiting the slowest transport rate was halted after 1.5 pore volumes had been exchanged. The second SDU Cell 2A sample continues to be evaluated and approximately 8 pore volumes have been exchanged for the sample. [SRRA099188-000005]

The Tc-99 concentrations and cumulative Tc-99 release for the SDU Cell 2A samples are generally higher than previously observed for the Tc-spiked, simulant saltstone. The range of Tc-99 concentrations measured for the Tc-spiked sample is 4.9E-09 mol/L to 3.6E-07 mol/L; in comparison, the Tc-99 concentrations for the two SDU Cell 2A samples has ranged 4.1E-08 mol/L to 5.0E-07 mol/L. In contrast, the concentration of nitrate in leachates of all three samples is much more consistent; both SDU Cell 2A samples and the Tc-spiked sample indicate an initial concentration of around 1.0 mol/L that subsequently decreases gradually to less than 0.4 mol/L. At the time of test cessation, the SDU Cell 2A with low flow rate indicated the following key values: NO_3^-, Tc-99, and Cs-137 concentrations of 0.37 mol/L, 9.5E-08 mol/L, and 2.3E-09 mol/L and cumulative percentages leached of 34.5%, 4.8%, and 6.7%, respectively. The final SHC was measured at 6.6E-11 cm/sec. After the exchange of 8 pore volumes, the second SDU Cell 2A sample indicates NO_3^-, Tc-99, and Cs-137 concentrations of 0.02 mol/L, 4.1E-08 mol/L, and 2.5E-10 mol/L and cumulative percentages leached of 77.4%, 11.7%, and 11.7%, respectively. The December 2017 measured SHC for this sample is 5.3E-10 cm/sec. [SRRA099188-000005]

The DLM samples described thus far have been evaluated on the originally developed DLM system in which a constant pressure is set and the flow rate allowed to vary based on changes with respect to the sample SHC. The newly designed system utilizes mechanical pumps in which a constant flow rate through the sample can be set. It should be noted that this new system has not performed as intended since the flow rates continue to vary based on changes in sample SHC. The original intent for being able to control the flow rate was two-fold. Firstly, the DLM experiments as currently designed are slow as indicated by the transport of only 15 pore volumes in 27 months for the aforementioned Tc-spiked sample. Secondly, it was envisaged that being able to control the flow rate would enable the impact of permeant residence time to be determined. To date, four samples have been evaluated on this revised DLM system and while the desired flow rates were set between 0.2 mL/day and 1.5 mL/day all samples are currently operating at rates at or below

0.1 mL/day. These samples have had less than a pore volume exchanged and as such it is too early to make any determinations with respect to sample-to-sample similarities or differences. In comparison, to the aforementioned Tc-spiked and SDU Cell 2A samples, the Tc-99 concentrations in the leachates is high at a measured value of 1.0E-06 mol/L. However, with leachate pH values in the range of 11.8 to 12.8 these concentrations are generally aligned with the thermodynamically-predicted solubility limits for reduced $TcO_2.xH_2O$ phases.

In FY2018, DLM testing continued for one of the SDU 2A core samples and to date has undergone approximately 12.5 pore flushes since the beginning of the experiment in March 2016. Initially the concentration of Tc-99 in the SDU 2A core leachate remained fairly consistent around a value of 2E-07 mol/L. Once the pH of the leachate dropped below 11, the concentration of Tc-99 in the leachate correspondingly dropped to around 5E-08 mol/L where it has remained since. The consistency of the Tc-99 concentrations measured in the leachate, both before and after the pH drops below 11, supports the idea that Tc-99 release from saltstone is initially solubility controlled. [SRRA099188-000003]

Four new DLM experiments were begun in FY2018 on saltstone simulants synthesized using a 45:45:10 dry feed composition (i.e., 45% Lehigh BFS, 45% fly ash, and 10% ordinary Portland cement) and the ARP-MCU simulant salt solution. Two of the saltstone simulants were spiked with Tc-99 while the other two were spiked with I-129. The two Tc-99 spiked saltstone simulants were found to have higher initial Tc-99 concentrations in the leachate as compared with the SDU 2A core studied in FY2018. Similar to the SDU 2A core, a decrease in the Tc-99 leachate concentration was observed for one of the saltstone simulants once the pH drops below 11. Comparison of the DLM data collected from both the Tc-99 spiked saltstone simulants and SDU 2A core, with theoretical solubility curves for Tc solid phases, suggests that under reducing conditions Tc release from saltstone is solubility-controlled by one or more hydrated Tc(IV)-oxide solid phases ($TcO_2 \cdot x(H_2O)$). [SRRA099188-000003] In addition, the concentration data provided from these DLM experiments allowed for the solubility values used in modeling Tc-99 release from saltstone to be updated for implementation in the FY2019 SDF PA. [SRR-CWDA-2018-00046; SRR-CWDA-2019-00001]

As mentioned previously, two of the saltstone simulants used in the FY2018 DLM experiments were spiked with I-129. The leachate data collected from these experiments suggests that I-129 is retained to some extent in the saltstone matrix (i.e., I-129 has a non-zero K_d with respect to saltstone). [SRRA099188-000005] This retention was later quantified by means of an iodine K_d value estimated using a GoldSim-based optimization model. [SRR-CWDA-2018-00045] The K_d value derived from this approach was subsequently implemented into the FY2019 SDF PA. [SRR-CWDA-2019-00001]

In February of 2019, four new DLM experiments began. These new DLM experiments utilized saltstone simulants spiked with both Tc-99 and I-129, and all were prepared using Lehigh BFS to better understand what impact the new slag has on saltstone's hydraulic (i.e., SHC) and transport (i.e., release rates) properties. Two different dry-feed formulations were used in synthesizing the new saltstone simulants: the historically used 45:45:10 mix (i.e., 45 wt% BFS, 45 wt% FA, and 10 wt% OPC) and a 60:40 cement-free mix (i.e., 60 wt% BFS, 40 wt% FA). Both simulants were prepared using a salt solution simulant representative of Tank 50's chemical composition once the Salt Waste Processing Facility (SWPF) begins operations. [SRR-CWDA-2018-00033]

Results to date from the FY2019 DLM experiments suggest that the 60:40 cement-free formulation is comparable in performance, from a PA perspective, to the historically used 45:45:10 saltstone. These experimental results, in conjunction with fresh property testing performed by System One (2019-07V1JE4009-0001), are being used to inform the liquid waste contractor's decision regarding implementation of a cement-free saltstone and its ramifications from a PA perspective. It should be noted that at this time no decision has been made. Once a final decision has been reached, a UWMQE will be performed as part of the overall technical evaluation if the liquid waste contractor elects to move forward with the cement-free formulation.

Plans for FY2020 center around three specific categories: 1) continuation of the FY2019 DLM experiments along with the addition of a new SDU 2A core DLM experiment, 2) saltstone degradation over time, and 3) the effects of curing time on saltstone properties relevant to the PA.

6.2 Long-Term Radiological Lysimeter Program

As part of a multi-year evaluation of the long-term behavior of radionuclides in the tank farms and saltstone disposal facility (SDF), two reports were issued in September 2018. The first report (SRRA021685-000011) documented concentrations measured in field lysimeter effluents from the fourth quarter of FY2017 and the second quarter of FY2018.

Key findings from this report include:

- Lysimeters containing NpO_2NO_3(s) sources (Np in the +5 oxidation state) continue to show measurable Np breakthrough. Starting in FY2016, Np was observed in the effluent of lysimeter 32 which contains a relatively insoluble NpO_2(s) source. The observation of Np in the effluent from this lysimeter implies that the NpO_2(s) is becoming oxidized and releasing Np(V) which can transport through the lysimeter with a relatively low K_d.
- Co-60 was only measured in the effluent of cement sources (lysimeters 4 and 5) for samples analyzed in FY2018. From the historical lysimeter data, it appears that Co-60 mobilization is highest in cement sources with no BFS, followed by saltstone sources, and lastly sediment/filter paper sources. It should be noted that in these experiments all cement and saltstone lysimeter sources contained higher concentrations of Co-60 in the effluent relative to a control with the gamma suite of radioisotopes added directly to a filter (i.e., sediment/filter paper sources). It is unclear what is causing this enhanced mobility of a small fraction of Co-60 in the cement and saltstone sources.

The second report (SRRA021685-000010) documented the detailed solid phase analysis of a field lysimeter (lysimeter 41) with an emplaced $Pu(V)NH_4(CO_3)$(s) source.

Key findings from this report include:

- Consistent with previous lysimeter studies, both downward and upward migration of Pu was observed. A lower degree of upward migration is proposed to be due to the shorter time for Pu diffusion in the current lysimeter (2 years) compared with previous lysimeter studies (11 years).
- The downward transport of Pu in the current $Pu(V)NH_4(CO_3)$(s) lysimeter was greater than that observed for previous $PuCl_3$, $Pu(NO_3)_4$, and $Pu(C_2O_4)_2$ bearing lysimeters but very similar to the PuO_2(s) lysimeter (lysimeter 44) analyzed in FY2017. [SRRA021685-

000008] Researchers have proposed multiple working hypotheses to explain the enhanced transport observed for the $Pu(V)NH_4(CO_3)(s)$ and $PuO_2(s)$ lysimeters.

- Working Hypothesis #1: Transport of Pu as $PuO_2(s)$ colloids. The $Pu(V)NH_4(CO_3)(s)$ source transforms to a $PuO_2(s)$ phase similar to the one found in lysimeter 44 (colloidal $PuO_2(s)$ source, SRRA021685-000008). The Pu colloids allow for enhanced transport of the radionuclide.
- Working Hypothesis #2: Transport is due to differing solubility values in the Pu source material. Evidence from the literature strongly suggests that the oxidized Pu present in $Pu(V)NH_4(CO_3)(s)$ will rapidly reduce to Pu(+IV), perhaps as a $PuO_2(s)$ phase similar to that found in lysimeter 44 (colloidal $PuO_2(s)$ source, SRRA021685-000008). Some of the Pu(+IV) present in the $PuO_2(s)$ phase oxidizes over time to the more mobile Pu(+V) oxidation state and is transported a short distance through the soil prior to being re-reduced and once again forming a $PuO_2(s)$ phase. The plutonium continues to undergo cycles of re-oxidation followed by re-reduction thereby allowing it to slowly traverse down the lysimeter column. This mechanism is analogous to the one proposed for mobilization of Np observed in the $NpO_2(s)$ lysimeter (lysimeter 32, SRRA021685-000011).

- Desorption experiments using Pu contaminated soils retrieved from this lysimeter indicated conditional desorption distribution coefficients of log K = 3.2 ± 0.2 mL/g. There was no apparent difference between unfiltered and ultra-filtered samples during the desorption experiments indicating that either 1) colloids are not present in these samples as hypothesized or 2) colloids sorb strongly to the soil and do not desorb.

Lysimeter effluent testing in conjunction with solid phase analysis of the lysimeter cores and source material provides researchers with a robust data set specific to the SRS that can provide less ambiguous assignment of transport mechanisms, including K_d values, and bolster confidence in PA modeling assumptions. For example, values taken from the lysimeter solid phase analysis study provide strong support for vadose zone plutonium K_ds that are one to two orders of magnitude higher than what is currently used in PA modeling at SRS.

In FY2019, eleven new lysimeters, containing plutonium, neptunium, or radium sources were installed at the Radionuclide Field Lysimeter Experiment (RadFLEx) facility at SRS.

Additional work discussed below was performed in FY2019 but due to delays in sample shipments from SRS to Clemson University, the results of these studies were not available as of October 1, 2019. Once all analyses are finished, the results will be documented in Clemson's annual end of the fiscal year report. Additional work performed in FY2019 includes:

- Analyses of field lysimeter effluents from the fourth quarter of FY2018 and the second quarter of FY2019.
- Desorption experiments conducted on gamma-emitting isotopes (i.e., Cs-137, Co-60, Ba-133, Eu-152) sorbed to SRS soil. This work will provide K_d values that are directly applicable to PA modeling at SRS.

- Low level measurements and oxidation state analysis of plutonium present in lysimeter effluent. These measurements will help support or disprove the colloidal transport of Pu in SRS sediment.

In FY2020, effluent samples will continue to be collected quarterly from the field lysimeters and transported to Clemson University for analysis. Monthly sampling and analysis will be performed for lysimeters where radionuclides have previously been detected in the leachate (e.g., Lysimeter 30 and Lysimeter 32 which utilize a Np(V) and a Np(IV) source, respectively). In addition, three lysimeters containing new sources will also be installed in FY2020. One lysimeter will possess a radium source placed directly in soil while the other two lysimeters will each contain a saltstone puck that has been spiked with iodine.

7. PLANNED OR CONTEMPLATED CHANGES

Sections 7.1 through 7.7 discuss planned work that are part of PA maintenance and monitoring activities. The DOE has performed a number of additional activities to support a 2019 revision to the SDF PA. The FY2019 SDF PA includes model revisions, UWMQE analysis recommendations (i.e., GSA model updates), lessons learned, and incorporation of the latest input values as developed through ongoing studies (see Sections 6 and 7.6), as well as other recent literature reviews and analyses, as appropriate.

7.1 Revise the Closure Plan

As stipulated in the SDF DAS (WDPD-12-49), and as part of the PA maintenance program, the SDF Closure Plan is reviewed annually to determine if additional revision is required. If a revision is required, then an update to the SDF closure plan will be submitted to DOE for approval. The SDF Closure Plan was reviewed and found to be consistent with current SDF facility and operational conditions for FY2019. An update of the SDF Closure Plan will be necessary in FY2020 after approval of the FY2019 SDF PA revision due to changes in future SDU locations.

7.2 Revise the SDF Monitoring Plan

As stipulated in the SDF DAS (WDPD-12-49), and as part of the PA maintenance program, the SDF Monitoring Plan is reviewed annually to determine if additional revision is required. If a revision is required, then an update to the SDF Monitoring Plan will be submitted to DOE for approval. The SDF Monitoring Plan was reviewed and found to be consistent with current SDF facility and operational conditions for FY2019. An update of the SDF Monitoring Plan will be necessary in FY2020 after approval of the FY2019 SDF PA revision due to changes in future SDU locations.

7.3 Special Analyses

As discussed in Section 2.1.1, the latest SDF SA (FY2016 SDF SA) was issued in the first quarter of FY2017. A SA or PA revision was recommended to address the potential impacts of the 2016 GSA model updates per SRR-UWMQE-2017-00004; these impacts will be addressed as part of the FY2019 SDF PA revision rather than a SA. Section 7.5 discusses the decision to create a SDF PA revision.

7.4 Unreviewed Waste Management Question Evaluations

A formal system to evaluate disposal practice changes and proposed actions is in place at the SDF. The process consists of providing screening and if necessary UWMQEs of proposed activities and new information. The Unreviewed Waste Management Question (UWMQ) process will continue to be required throughout the life of the facility. This process is implemented via Manual S4 Procedure ENG.46 Revision 4.

7.5 Revise the Performance Assessment

In FY2019 a revision to the SDF PA was prepared (SRR-CWDA-2019-00001). The timing of this PA revision is associated with four main drivers:

1. The design and layout of the SDF Saltstone Disposal Units (SDUs) have undergone major changes since the last PA revision.
2. The breadth of research and development activities in recent years has provided new information and increased the confidence in key transport modeling inputs and assumptions.
3. Three Special Analyses have been conducted since the 2009 PA and this information needed to be consolidated into the new PA revision.
4. DOE-STD-5002-2017 states that PAs should be revised at a minimum every ten years and the previous SDF PA was completed in 2009.

The latest draft of the 2019 SDF PA (Revision B) is currently under review by the United States Department of Energy Low Level Waste Disposal Facility Federal Review Group (LFRG).

Once the LFRG review is complete and any key issues have been resolved, the updated PA will be issued. As part of PA implementation, related documents may be updated to ensure that any performance-affecting assumptions or requirements within the updated PA will be protected. Such documents may include, for example, the *Unreviewed Waste Management Question Requirements Document for Saltstone Facility* (SRR-CWDA-2011-00196), the *Waste Acceptance Criteria for Transfers to the Z-Area Saltstone Production Facility During Salt Disposition Integration (SDI)* (X-SD-Z-00004), and the *Closure Plan for the Z-Area Saltstone Disposal Facility* (SRR-CWDA-2013-00037).

7.6 Studies

PA-related testing and research activities are being performed as part of the on-going maintenance activities aimed at reducing uncertainty in the 2009 SDF PA model or are verification sampling and analysis of materials properties used in the 2009 SDF PA. As ongoing research provides new information or reduces uncertainty, this information will be evaluated (via the UWMQ and SA process) against the information used as a basis for the 2009 SDF PA modeling. Once the FY2019 SDF PA revision is completed, any new information will be evaluated (via the UWMQ and SA process) against the information used as a basis for the 2019 SDF PA modeling.

Below is a brief list of testing and research activities currently planned for FY2020.

- Radionuclide Leaching Characteristics from Saltstone Monolith (Simulated and Actual SDU Samples)

- Long-term Radiological Lysimeter Program
- Performance Assessment Monitoring

7.7 Performance Assessment Monitoring

Per the requirements in the DAS issued for the SDF (WDPD-12-49), a monitoring plan shall be written, approved, and implemented within one year of issuance of the DAS and updated at least every five years. This monitoring plan includes annual data review and evaluation. Following this annual data review and evaluation, any modifications to this monitoring plan that may be applicable will be noted and the plan updated, as necessary.

8. STATUS OF DAS CONDITIONS, KEY AND SECONDARY ISSUES

In FY2019 there were no DAS conditions in effect to report and there are no open key or secondary issues from any LFRG reviews. FY2019 SDF operations comply with the DAS.

9. COMPOSITE ANALYSIS SUMMARY

The annual evaluation of the SRS CA is covered by a separate report prepared by the Savannah River National Laboratory (SRNL). The latest evaluation as of the issuance of this review was issued in April 2019 for FY2018 (SRNS-RP-2019-00051).

10. CERTIFICATION OF CONTINUED ADEQUACY OF THE PA, CA, DAS, AND RWMB AND CONCLUSION

The current performance evaluation conducted on SDU 1, SDU 4, SDU Cells 2A/2B, SDU Cells 3A/3B, SDU Cells 5A/5B, and SDU 6 indicates SDF operations through FY2019 were within the performance expectation of the 2009 SDF PA, the FY2013 SDF SA, FY2014 SDF SA, and FY2016 SDF SA and comply with the DAS, the RWMB, and DOE O 435.1 requirements.

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