Technical Evaluation Report

Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility at the Idaho National Laboratory Site

Final Report

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TABLE OF CONTENTS

SECTION PAGE

LIST OF FIGURES

Page

LIST OF TABLES

Page

ABBREVIATIONS AND ACRONYMS

ITEM	MEANING
1-D	One-dimensional
3-D	Three-dimensional
Ac	Actinium
ADAMS	NRC Agencywide Documents Access and Management System
Ag	Silver
Al	Aluminum
Am	Americium
Be	Beryllium
Bi	Bismuth
BLR	Big Lost River
Bq	Becquerel
C	Carbon
°C	Degree Celsius
CA	Composite Analysis
CC	Clarifying Comment
Cd	Cadmium
CDF	Cumulative distribution function
Ce	Cerium
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
Cf	Californium
CFR	Code of Federal Regulations
Ci	Curie
Cm	Curium
cm	Centimeter
CNWRA®	Center for Nuclear Waste Regulatory Analyses
Co	Cobalt
Cs	Cesium
CSSF	Calcined Solids Storage Facility
DOE	U.S. Department of Energy
DOE-ID	U.S. Department of Energy, Idaho Operations Office
EDF	Engineering design file
EPA	U.S. Environmental Protection Agency
Eu	Europium
°F	Degree Fahrenheit
FEPs	Features, events, and processes
ft	Foot
Gd	Gadolinium
GSD	Geometric standard deviation
H-3	Tritium
HLW	High-level radioactive waste
Ho	Holmium
HPM	Historical Processing Model
hr	Hour
HRR	Highly radioactive radionuclide
HTD	Hard-to-detect
HWMA	Hazardous Waste Management Act
I	Iodine

ICP	Idaho Cleanup Project
in	Inch
In	Indium
INEEL	Idaho National Engineering and Environmental Laboratory
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
Kd	Distribution coefficient
kg	Kilogram
km	Kilometer
Kr	Krypton
L	Liter
La	Lanthanum
lbm	Pound-mass
LLW	Low-level radioactive waste
m	Meter
MCM	Mixing Cell Model
MEP	Maximum extent practical
MFAT	Multiple-factors-at-a-time
mi	Mile
MMPD	Mass mean particle diameter
mrem	Millirem
mSv	Millisievert
N/A	Not available
Nb	Niobium
Nd	Neodymium
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
Ni	Nickel
Np	Neptunium
NR	Not reported
NRC	U.S. Nuclear Regulatory Commission
NWCF	New Waste Calcining Facility
OFAT	One-factor-at-a-time
ORIII	Oxidized Region III
ORIGEN2	Oak Ridge Isotope Generation and Depletion Code
OU	Operable unit
Pa	Protactinium
PA	Performance assessment
Pb	Lead
pCi	Picocurie
Pd	Palladium
PDCF	Pathway dose conversion factor
pH	Measure of acidity (negative log of the hydrogen ion concentration)
Pm	Promethium
Pu	Plutonium
Ra	Radium
RADCAT	Risk and Dose Calcine Assessment Tool
RAI	Request for additional information
Rb	Rubidium
RCRA	Resource Conservation and Recovery Act
rem	Unit of dose equivalent

RI/BRA	Remedial investigation/baseline risk assessment
Ru	Ruthenium
RWMC	Radioactive Waste Management Complex
s	Second
Sb	Antimony
SBW	Sodium-bearing waste
Se	Selenium
Sm	Samarium
Sn	Tin
SNF	Spent nuclear fuel
SOF	Sum of fractions
Sr	Strontium
SRM	Staff Requirements Memorandum
SRPA	Snake River Plain Aquifer
SRS	Savannah River Site
SS	Stainless steel
Sv	Sievert
Tc	Technetium
Te	Tellurium
TEDE	Total effective dose equivalent
TER	Technical Evaluation Report
TFF	Tank Farm Facility
Th	Thorium
U	Uranium
WD	Waste Determination
WIPP	Waste Isolation Pilot Plant
WIR	Waste incidental to reprocessing
WMA	Waste Management Area
yr	Year
Zr	Zirconium

EXECUTIVE SUMMARY

On October 20, 2023, the United States Department of Energy (DOE) submitted the “Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility at the Idaho National Laboratory Site” for the U.S. Nuclear Regulatory Commission (NRC) to review. The DOE submittal to the NRC is required under Section 3116(a) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA). Section 3116 of the NDAA requires DOE to consult with the NRC when determining that certain wastes associated with spent fuel reprocessing are not high-level wastes (HLW). The draft waste determination addresses Calcined Solids Storage Facility (CSSF) bins (including integral equipment), transport lines, and residual waste. This Technical Evaluation Report (TER) presents information on DOE’s disposal strategy, the applicable review criteria, and NRC staff’s review approach, as well as NRC staff’s analysis and conclusions with respect to whether there is reasonable assurance that DOE’s proposed approach can meet the applicable NDAA criteria.

Based on the information provided by DOE, NRC staff has concluded in this TER that there is reasonable assurance that the applicable criteria of the NDAA can be met for residual waste associated with the CSSF. The NDAA also requires NRC, in coordination with the State of Idaho, to monitor DOE disposal actions to assess DOE compliance with the performance objectives in Title 10, *Code of Federal Regulations*, Part 61 (10 CFR Part 61), Subpart C. During its review of DOE’s draft waste determination, NRC identified factors for NRC to consider when monitoring DOE disposal actions to assess compliance with 10 CFR Part 61, Subpart C (see Section 4.6).

There are seven calcined solids storage facility bin sets, although one bin set was never operational and stores no waste. The six operational bin sets contain a total of 4,440 m^3 [1.57×10^5 ft^3] of calcined waste. Waste has not yet been retrieved from any of the six bin sets. Closure activities include waste retrieval to the maximum extent practical, and stabilization of the waste with a cementitious waste form. The assumed volume of residual waste is 5 cm [2 in], which is the basis for the long-term dose estimates provided in a performance assessment submitted with the draft waste determination. DOE plans to use a dry pneumatic vacuum retrieval and transfer system to remove residual waste from the bin sets. Studies and mockup testing is used to provide support for the assumption that the retrieval system will remove over 99 percent of the total volume of waste stored in the bin sets prior to closure.

The NDAA contains three criteria for determining that waste is not HLW. The first is that the waste does not require permanent isolation in a deep geologic repository for spent nuclear fuel (SNF) or HLW. This criterion allows for the consideration that waste may require disposal in a deep geologic repository even though the other criteria of the NDAA can be met. Consideration could be given to those circumstances under which geologic disposal is warranted to protect public health and safety and the environment (e.g., unique radiological properties of the waste). Because DOE has demonstrated that it can meet the other criteria in the NDAA, including the performance objectives in 10 CFR Part 61, Subpart C, and because there appears to be no other properties of the waste that would require deep geologic disposal, the NRC staff has reasonable assurance that NDAA Criterion One can be met.

The second criterion of the NDAA is that waste has had highly radioactive radionuclides (HRRs) removed to the maximum extent practical. To assess compliance with Criterion Two, the NRC staff assessed DOE Idaho’s estimated waste inventory, identification of HRRs, selection of treatment technology, and demonstration of removal to the maximum extent practical including consideration of the costs and benefits of additional radionuclide removal. NRC staff has reasonable assurance that DOE can meet Criterion Two that HRRs will be removed to the maximum extent practical based on staff's evaluation of DOE's selection of HRRs, DOE's selection and performance testing of its dry pneumatic vacuum retrieval and transfer system, and consideration of the costs and benefits of additional removal.

The third criterion of the NDAA is that waste will be disposed of in compliance with 10 CFR Part 61, Subpart C, performance objectives. Subpart C to 10 CFR Part 61 provides requirements for protection of the public, the inadvertent intruder, and individuals during operations; and also provides for site stability. To assess compliance with Criterion Three, NRC staff evaluated DOE's performance assessment, which provided long-term dose estimates for members of the public (i.e., the general population) from releases of radioactivity and dose to potential inadvertent intruders to evaluate performance objectives 10 CFR 61.41 and 61.42, respectively. NRC staff also evaluated DOE's draft waste determination, which provided information about DOE's radiation protection program to evaluate protection of individuals during operations (10 CFR 61.43), as well as provided information about stability of the disposal site after closure (10 CFR 61.44).

NRC staff has reasonable assurance that DOE can meet Criterion Three of the NDAA based on the information provided by DOE in its draft waste determination, performance assessment and responses to requests for additional information (RAIs), if certain key assumptions are verified during monitoring. NRC staff expects the dose to members of the public from releases of radioactivity from the facility to be below the 0.25 mSv/yr [25 mrem/yr] dose standard specified in 10 CFR 61.41, and that DOE can meet the 10 CFR 61.42 performance objective related to protection of individuals from inadvertent intrusion. Therefore, NRC staff has reasonable assurance that the 10 CFR 61.41 and 61.42 performance objectives can be met. Workers are protected by DOE regulations that are comparable to 10 CFR Part 20, and DOE controls are also in place to protect members of the public during operations. Therefore, NRC staff has reasonable assurance that DOE can meet the 10 CFR 61.43 requirements for protection of individuals during operations. DOE's closure plans, which include filling the bin sets and vaults with grout to provide structural stability and limit waste dispersal after HRRs have been removed to the maximum extent practical, provided sufficient information for NRC staff to have reasonable assurance that the 10 CFR 61.44 performance objective related to site stability could also be met.

For a broader and more detailed discussion of DOE's approach and NRC staff's analysis and conclusions, please see the relevant portions of the TER. All of the conclusions reached by NRC staff are based on DOE's draft waste determination submitted on October 20, 2023; DOE's responses to RAIs submitted in March 2025; supporting references; and information provided during meetings between DOE and NRC. If in the future, DOE determines it is necessary to revise its assumptions, analysis, design, or waste management approach and those changes are important to meeting the criteria of the NDAA, DOE should engage with the NRC. Note that NRC is providing consultation to DOE as required by the NDAA, and NRC is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW. This NRC staff assessment is a site-specific evaluation and is not a precedent for any future decision regarding non-HLW or incidental waste determinations at INL or other DOE sites.

1.0 INTRODUCTION

The U.S. Department of Energy (DOE) manages high-level waste (HLW) at sites across the DOE complex. From time to time, DOE may determine that certain waste resulting from reprocessing spent nuclear fuel (SNF) can be managed as low-level waste (LLW) [i.e., waste incidental to reprocessing (WIR)], rather than managed as HLW.

The NRC has a non-regulatory role in WIR as defined in Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA). The NDAA covers the DOE sites in Idaho and South Carolina (i.e., NDAA-Covered States). The NRC has two functions under the NDAA. Under NDAA Section 3116(a), the DOE must consult with the NRC prior to making the final waste determination. Under NDAA Section 3116(b), following the Secretary of Energy's final determination that the waste is WIR, the NRC monitors the DOE disposal actions in coordination with the NDAA-Covered State. NRC and the NDAA-Covered State assess the DOE disposal actions to determine compliance with the performance objectives set forth in Subpart C of Title 10, Part 61, of the *Code of Federal Regulations* (10 CFR Part 61), "Licensing Requirements for Land Disposal of Radioactive Waste."

The DOE issued the Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility (CSSF) at the Idaho National Laboratory (INL) site. This Draft CSSF 3116 Basis Document addresses residual calcine and CSSF structures and components that have had contact with calcine (DOE-ID, 2022). These structures and components include: (1) the waste storage and transfer equipment that comprises the bins, distributor lines, cyclones, and transport lines, (2) off-gas system, access risers, and rod-out lines, and (3) components contained within the bins such as the thermowells and corrosion coupons.

1.1 Facility and Site Description

1.1.1 Facility Description

The CSSF is located at the Idaho Nuclear Technology and Engineering Center (INTEC) on the INL Site. The INL is an approximately 2,300 km^2 [890 mi^2] reservation owned by the United States government and located in southeastern Idaho. The INTEC facility is located approximately 29 km [18 mi] from the closest eastern boundary, approximately 23 km [14 mi] from the closest western boundary, approximately 16 km [10 mi] from the closest southern boundary, and approximately 29 km [18 mi] from the closest northern boundary. INTEC's mission is to receive and temporarily store SNF and other radioactive waste and remediate legacy waste (DOE, 2023). The CSSF stores solid HLW (referred to as "calcine") and other non-radioactive material (primarily startup bed nonradioactive material) in stainless steel (SS) bins housed in six discrete reinforced-concrete vaults, known as CSSFs 1 through 6, each containing 3 to 12 SS storage bins (Figure 1-1). Calcine was generated from 1963 to 2000 by converting (calcining) liquid HLW from the reprocessing of SNF and non-reprocessing waste stored in tanks at the INTEC Tank Farm Facility into a granular solid (i.e., calcine). The liquid HLW resulted from reprocessing of SNF by DOE and its predecessor agencies from 1952 to 1992 at INTEC.

Calcine production began in 1963 at the Waste Calcining Facility (CSSFs 1-3). Operations switched to the New Waste Calcining Facility in 1982 (CSSFs 4-6). CSSF 7 was built but never placed into service. CSSFs 1 through 5 are nearly filled to capacity, while CSSF 6 is partially

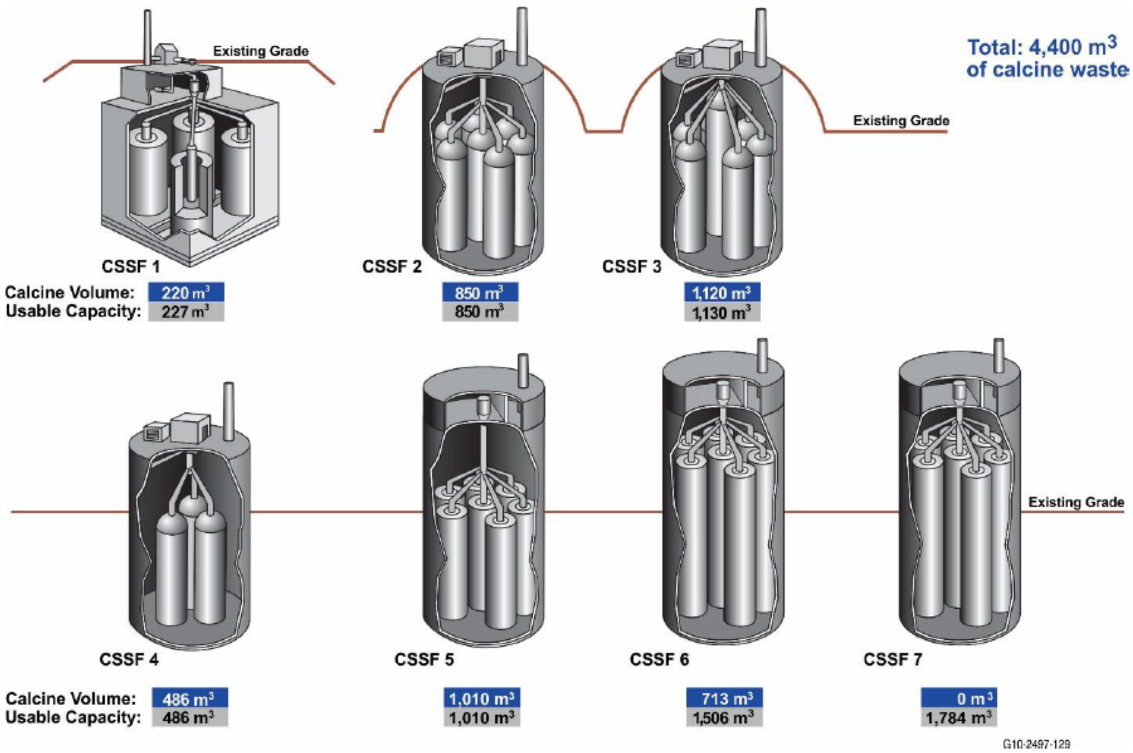


Figure 1-1: Calcined Solid Facility Bin Sets and Stored Volumes [from Figure 1-3 of DOE-ID (2022)]

filled. Calcining operations ended in 2000 with approximately 4,440 m^3 [1.57×10^5 ft^3] of calcine produced and stored in the six operational bin sets (DOE, 2023). Additional information about the CSSF can be found in Section 2.11 of DOE (2023).

1.1.2 Site Description

A description of the site including land and water use resources; terrestrial and aquatic biota; local meteorology and climatology; geology, hydrology, and hydrogeology is found in Chapter 2 of the CSSF performance assessment (PA). A short summary is provided below. Please see the PA for additional information. The site is described as a relatively flat, semiarid, sagebrush desert with elevations ranging from 1,460 m [4,790 ft] to the south to 1,802 m [5,912 ft] to the northeast with an average elevation of 1,500 m [5,000 ft] above sea level. The population surrounding the INL site affected by site activities include employees, nearby residential populations, and ranchers who graze livestock (cattle and sheep) and hunters (elk and pronghorn) on or near the site. The Shoshone-Bannock Tribes of the Fort Hall Indian Reservation has access to areas of cultural and religious significance at the INL site including the Middle Butte area.

Prevailing wind directions are southwesterly. Average monthly temperatures for a 65-year record at the Central Facilities Area range from a low of −8.7 °C [16.3 °F] in January to a high of 20.3 °C [68.5 °F] during July. Extreme temperatures have been recorded as high as 40.6 °C [105 °F] in July and as low as −43.9 °C [−47 °F] in December. The average annual precipitation is 21.3 cm [8.3 in]; the highest recorded annual amount of precipitation was 36.6 cm [14.4 in] and the lowest amount was 7.72 cm [3.04 in]. The minimum and maximum annual snowfalls are 17.3 and 151.6 cm [6.8 to 59.7 in], respectively, with an annual average of 65.8 cm [25.9 in] and a maximum average monthly of 15.5 cm [6.1 in], occurring in January. Potential evaporation rates are relatively high with a large fraction of the precipitation rate potentially subject to evapotranspiration; however, precipitation is most likely to infiltrate the ground during late winter to spring because of the low evapotranspiration rates during this time period and particularly associated with spring snow melt when the ground is no longer frozen.

Regarding the geology of the site, greater than 121 basalt flow groups and 102 sedimentary interbeds consisting of clay, silt, sand and gravel underlie the INL site above the effective base of the Snake River Plan Aquifer. Individual basalt flows range in thickness from 3 to 15 m [10 to 50 ft] and are locally interbedded with scoria and layers of sediment that are up to 15 m [50 ft] thick. Surficial alluvium composed of alluvial, fluvial, and eolian silt, sand and gravel are deposited on top of the uppermost basalt flow and range in thickness from 6.7 to 18.6 m [22 to 61 ft]. Interbeds of hydrogeological significance are listed in Section 2.1.6.1.1 of the PA with Figure 2-8 providing a north-south geological cross section at the INTEC illustrating an approximately 143 m [470 ft] vadose zone based on 2018 water level measurements from wells at INTEC. Six INTEC lithological marker units are listed as follows (see page 2-17 for descriptions of each unit):

1. Surficial alluvium
2. 34-m [110-ft] interbed
3. High K_2O basalt flow
4. 43-m [140-ft] interbed
5. Middle massive basalt flow
6. 116-m [380-ft] interbed

Information on Seismology and Volcanology is found in Sections 2.1.6.2 and 2.1.6.3, respectively. Hazard assessments, which include estimates of peak ground motions, were completed for all facility areas at INL. Design criteria for the CSSF bin sets are based on design codes, standards, regulations, and DOE orders existing at the time of construction. The PA indicates that the CSSF storage vaults are sufficiently robust to expect no radiological release in the event of a Performance Category 3 earthquake. The PA also indicates that renewed explosive volcanism at the INL site is very unlikely. Data suggest that the INL site will remain an area of subsidence and net deposition (erosion is not expected to be significant), although climate fluctuations could influence sedimentation patterns.

The Big Lost River (BLR) experiences intermittent flow (regulated by controlled releases from Mackay Reservoir and is also dependent on winter snowpack), but recharge is not expected to impact the CSSF even in times of flow. A flooding study showed minimal impacts from an extreme precipitation event and overtopping of the Mackay Dam. While one to two meters of water could cover INTEC for a short duration, the study predicts a small wetting front and shallow advancement into the alluvial soils (DOE-ID, 2003).

Water from snowmelt and heavy rains can infiltrate to depths where it cannot evaporate and will recharge perched water and the Snake River Plan Aquifer (SRPA). The combination of coarse surficial sediments and lack of vegetation permits infiltration of a large fraction of the natural precipitation. The objective of ongoing remedial actions under the Comprehensive Response, Compensation and Liabilities Act is to reduce infiltration and recharge rates over time including eliminating anthropogenic water leaks, landscape watering, steam condensate drip-leg discharges to the ground, as well as lining drainage ditches, and directing runoff from asphalted areas to lined ditches and evaporation ponds.

The SRPA underlying the INL site is one of the largest and most productive groundwater resources in the United States. At over 322 km [200 mi] long and 64 to 94 km [40 to 60 mi] wide, the SRPA runs northeast to southwest from Ashton, Idaho, to Bliss, Idaho, with boundaries formed by contacts with less permeable rocks at the margins of the eastern Snake River plain. Hydraulic conductivity in the fractured rock near INTEC commonly exceeds 300 m/day [1,000 ft/day], with a maximum value of 2,700 m/day [8,800 ft/day] and groundwater velocities of 1.5 m/day [5 ft/day] at the former INTEC injection well.

1.2 DOE-ID Calcined Solids Storage Facility Closure Strategy

1.3 Waste Determination Criteria

Since 1969, the concept of incidental waste or WIR has been recognized; certain waste can be managed based on its risk to human health and the environment, rather than the origin of the waste. Some waste that originate from reprocessing of SNF are highly radioactive and need to be treated and disposed of as HLW. Other reprocessing waste does not pose the same risk to human health and the environment and therefore does not need to be disposed of as HLW. DOE uses waste determinations to evaluate whether reprocessing waste is HLW or incidental waste.

The original incidental waste criteria were approved by NRC's Staff Requirements Memorandum (SRM) dated February 16, 1993, in response to SECY-92-391, "Denial of PRM 60-4–Petition for Rulemaking from the States of Washington and Oregon Regarding Classification of Radioactive Waste at Hanford." These criteria are described in the March 2, 1993, letter from R. Bernero, NRC, to J. Lytle, DOE as follows (NRC, 1993): (i) the waste has been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; (ii) the waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR Part 61; and (iii) the waste is to be managed, pursuant to the Atomic Energy Act, so that safety requirements comparable to the performance objectives set out in 10 CFR Part 61 are satisfied.

In October 2004, the NDAA was signed into law. NDAA Section 3116 allows DOE to continue to use a process to determine that waste is not HLW and requires that DOE consult with NRC on its non-HLW determinations. However, the NDAA is applicable only to South Carolina and Idaho and does not apply to waste transported out of those states. The NDAA establishes the following criteria for determining that waste is not HLW:

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

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

3. (A) does not exceed concentration limits for Class C low-level waste as set out in Section 61.55 of Title 10, *Code of Federal Regulations*, and will be disposed of—

    i. in compliance with the performance objectives set out in Subpart C of Part 61 of Title 10, *Code of Federal Regulations*; and

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

(B) exceeds concentration limits for Class C low-level waste as set out in Section 61.55 of Title 10, *Code of Federal Regulations*, but will be disposed of—

i. in compliance with the performance objectives set out in Subpart C of Part 61 of Title 10, *Code of Federal Regulations*;

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

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

1.4 NRC Review Approach

The NDAA requires (i) that DOE consult with NRC on its non-HLW determinations and (ii) that NRC, in coordination with the Covered State, monitor disposal actions taken by DOE for the purpose of assessing compliance with NRC regulations in 10 CFR Part 61, Subpart C. If the NRC considers any DOE disposal actions are not in compliance, NRC shall inform DOE, the covered State, and congressional subcommittees. In addition, the NDAA provides for judicial review of any failure of the NRC to carry out its monitoring responsibilities.

Prior to the NDAA, DOE has periodically requested NRC to provide a technical review of specific WIR determinations. NRC has provided technical assistance and advice to DOE regarding its WIR determinations but did not provide regulatory approval for DOE's actions. In past reviews, the staff reviewed DOE's WIR determinations to assess whether they had sound technical assumptions, analysis, and conclusions with regard to meeting the applicable incidental waste criteria. The staff typically evaluated information submitted by DOE, generated request for additional information questions (RAIs), met with DOE representatives to discuss technical questions and issues, and documented final review results in a technical evaluation report (TER). In December 2005, NRC completed its first waste determination technical evaluation under the NDAA for salt waste disposal at the Savannah River Site (SRS) and the review was completed in a similar manner to the waste determinations reviewed prior to the NDAA (NRC, 2005).

NRC staff's review, documented in this TER, was based on the DOE Idaho Operations Office (DOE-ID) "Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility at the Idaho National Laboratory Site" (DOE, 2023). A publicly available version of the draft waste determination was submitted by DOE-ID on October 20, 2023, along with references. The NRC staff performed a technical review of the information and sent an RAI to DOE-ID on April 8, 2024 (NRC, 2024a). The RAIs included questions about removal of key radionuclides to the maximum extent practical, engineered barrier and natural system performance, inadvertent intrusion, site stability, and included requests for clarification on various topics. NRC and DOE held public meetings on August 27, 2024 (NRC, 2024b; ML24270A086), September 10, 2024 (NRC, 2024c; ML24296A001) and September 24, 2024 (NRC, 2024d; ML25043A048) to discuss the NRC RAIs. In a letter dated March 27, 2025 (DOE-ID, 2025a), DOE-ID submitted its RAI responses and additional references. A public meeting was held June 26, 2025, for NRC and DOE to discuss the RAI responses. In an email dated July 28, 2025 (DOE-ID, 2025b), DOE responded to NRC staff questions raised during the public meeting.

NRC staff reviewed the draft waste determination and supporting documentation to assess whether it had sound technical assumptions, analyses, and conclusions with regard to meeting NDAA criteria, and that DOE's proposed closure of the CSSF protects public health and safety and the environment. This approach is consistent with that proposed by the NRC staff in SECY-05-0073, "Implementation of New U.S. Nuclear Regulatory Commission Responsibilities Under the National Defense Authorization Act of 2005 in Reviewing Waste Determinations for the U.S. Department of Energy," dated April 28, 2005, and approved by the Commission in the SRM dated June 30, 2005. This TER addresses each of the applicable criteria in the NDAA and presents the NRC staff's approach, assumptions, and conclusions, as well as identified key areas to be targeted for monitoring that are important to meet the performance objectives in 10 CFR Part 61, Subpart C.

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

1.5 Previous Waste Determination Reviews for INL

In 2001, DOE requested NRC consultation on two draft WIR determinations for INL. The first WIR determination involved sodium bearing waste (SBW) that would be removed from the HLW tanks and disposed of at DOE's Waste Isolation Pillot Plant (WIPP). Because that was transuranic waste and would be disposed of at a facility regulated by the U.S. Environmental Protection Agency (EPA), the NRC staff only reviewed whether DOE's methodology would meet the DOE Order, DOE O 435.1 ("Radioactive Waste Management"), criterion of being processed to remove key radionuclides to the maximum amount technically and economically practical (NRC, 2002a). The staff's conclusions were transmitted to DOE on August 2, 2002, and the staff stated that DOE's methodology appeared to meet the criterion.

The second WIR determination for INL involved the HLW tanks also located at the INTEC at INL. The staff used the two WIR criteria provided in the NRC's Final West Valley Policy Statement (NRC, 2002b) and concluded that DOE appeared to have reasonably analyzed the relevant considerations in concluding that the residual waste in the tanks could meet the two WIR criteria (NRC, 2003). DOE provided a waste determination for NRC review for the INTEC Tank Farm Facility (TFF) again in 2005 under the NDAA. While the previous NRC staff review provided risk insights, the NRC performed a second review and issued a second TER in 2006 updating the previous review (NRC, 2006) to specifically address the NDAA criteria, which are slightly different compared to the NRC's Final West Valley Policy Statement used in the first review and to address new information provided to NRC since the previous review.

2.0 CRITERION ONE

The waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW (NDAA). Waste Disposal Criterion One allows for the consideration that waste may require disposal in a geologic repository even though the two other criteria of the NDAA may be met. Consideration could be given to those circumstances under which geologic disposal is warranted in order to protect public health and safety and the environment, for example, unique radiological characteristics of waste or non-proliferation concerns for particular types of material.

2.1 NRC Review and Conclusions

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

3.0 CRITERION TWO

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

The NRC staff evaluated this criterion by analyzing DOE-ID's (i) methodology for developing radionuclide inventories for the bin sets and auxiliary equipment; (ii) process for identifying highly radioactive radionuclides; (iii) selection of waste treatment technology; and (iv) demonstration of removal to the maximum extent practical, including analysis of the costs and benefits of additional radionuclide removal. For the purpose of reviewing DOE waste determinations, NRC staff believe that highly radioactive radionuclides (HRRs) are those radionuclides that contribute most significantly to risk to public, workers, and the environment (NRC, 2007).

3.1 Waste Inventory

This section describes DOE's approach to determining radionuclide inventories for the bin sets (including nested bins, bin walls, stiffening rings, and other internal components) and auxiliary equipment. This section also briefly discusses uncertainty with respect to inventory estimates (e.g., uncertainty due to limited sampling of variable waste streams, modeling assumptions and uncertainty, use of process knowledge, volume estimates, and density measurements), although more detailed discussion is found in Section 4.2.8.

The inventory in each of the CSSF bins: (i) is used to demonstrate that the waste has had HRRs removed to the maximum extent practical; (ii) determines whether the waste is Class C; and (iii) is used to develop the source term in the PA.

3.1.1 Projected Calcine Bin Set Inventory

DOE attempted to develop a projected inventory that was reasonably conservative for the bin sets following operational closure. See Table 3-1 for final estimated inventory for each bin set. Because DOE has not yet performed retrieval operations, the inventory calculations were based on historical liquid sample data, total volume of liquid calcined, calciner operating data, and CSSF operating data. However, DOE notes that relatively few historical calcine samples with extensive analyses exist to use as the basis for calcine composition (EDF-11126; DOE-ID, 2021). As a result, DOE used large Microsoft® Excel® spreadsheets, referred to as the Historical Processing Model (HPM), to compile the available data and calculate projected radionuclide inventory at the time of closure. The HPM has been reviewed for accuracy and completeness and has been compared with calcine sample data, as described Section 2.11.3 of the Waste Determination (WD). When evaluating the composition of calcine, DOE separated each bin into segments, or sections. The bin segments do not correspond to changes in calcine composition or chemically distinct layers of calcine; a calcine segment may contain a single, uniform type of calcine, or it may contain multiple layers of chemically different calcine. Instead, the bin segments correspond to the location of level-indicating thermocouples in each bin. Across the six bin sets at the CSSF, 43 bins contain calcine, with 6 to 12 segments in each bin; the HPM calculates the average calcine composition for a total of 337 bin segments (Staiger and Swenson, 2021). For radionuclides that were not analyzed from waste sample data, the HPM uses ORIGEN2-based models to calculate the ratios of each radionuclide to Cs-137 for different waste types and campaigns.

Table 3-1: DOE-estimated Radionuclide Inventory (Ci) decayed to January 1, 2016, Based on Assumed 5.1 cm [2 in] Residual Calcine Depth Left After Retrieval

Radionuclide	CSSF 1	CSSF 2	CSSF3	CSSF 4	CSSF 5	CSSF 6	Total
Ac-227	1.06×10^{-4}	6.55×10^{-5}	5.01×10^{-6}	2.88×10^{-6}	6.19×10^{-6}	2.89×10^{-6}	1.89×10^{-4}
Ag-108m	6.52×10^{-8}	1.02×10^{-7}	1.42×10^{-7}	8.15×10^{-8}	1.75×10^{-7}	8.18×10^{-8}	6.48×10^{-7}
Am-241	1.19	4.99	8.07	5.06	9.48	2.29	3.11×10^{1}
Am-242m	1.70×10^{-4}	2.33×10^{-3}	5.01×10^{-3}	2.88×10^{-3}	6.19×10^{-3}	2.89×10^{-3}	1.95×10^{-2}
Am-243	8.40×10^{-5}	3.69×10^{-4}	1.03×10^{-3}	6.44×10^{-4}	1.13×10^{-3}	6.41×10^{-4}	3.90×10^{-3}
Be-10	1.65×10^{-6}	1.41×10^{-6}	9.63×10^{-7}	5.53×10^{-7}	1.19×10^{-6}	5.55×10^{-7}	6.33×10^{-6}
Bi-210m	4.09×10^{-22}	5.00×10^{-20}	1.12×10^{-19}	6.43×10^{-20}	1.38×10^{-19}	6.45×10^{-20}	4.29×10^{-19}
C-14	6.63×10^{-6}	3.95×10^{-6}	2.60×10^{-9}	1.49×10^{-9}	3.21×10^{-9}	1.50×10^{-9}	1.06×10^{-5}
Cd-113m	4.72×10^{-1}	5.38×10^{-1}	5.77×10^{-1}	3.31×10^{-1}	7.13×10^{-1}	3.33×10^{-1}	2.96
Ce-142[a]	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cf-249	2.18×10^{-16}	2.35×10^{-12}	5.28×10^{-12}	3.03×10^{-12}	6.53×10^{-12}	3.05×10^{-12}	2.02×10^{-11}
Cf-250	1.86×10^{-17}	1.10×10^{-12}	2.47×10^{-12}	1.42×10^{-12}	3.05×10^{-12}	1.42×10^{-12}	9.45×10^{-12}
Cf-251	5.82×10^{-19}	4.19×10^{-14}	9.41×10^{-14}	5.41×10^{-14}	1.16×10^{-13}	5.43×10^{-14}	3.61×10^{-13}
Cm-243	8.11×10^{-6}	2.82×10^{-4}	6.23×10^{-4}	3.58×10^{-4}	7.71×10^{-4}	3.60×10^{-4}	2.40×10^{-3}
Cm-244	1.25×10^{-4}	3.91×10^{-3}	8.04×10^{-3}	4.91×10^{-3}	9.20×10^{-3}	4.83×10^{-3}	3.10×10^{-2}
Cm-245	4.29×10^{-8}	1.64×10^{-6}	3.64×10^{-6}	2.09×10^{-6}	4.50×10^{-6}	2.10×10^{-6}	1.40×10^{-5}
Cm-246	9.78×10^{-10}	1.74×10^{-7}	3.91×10^{-7}	2.25×10^{-7}	4.83×10^{-7}	2.25×10^{-7}	1.50×10^{-6}
Cm-247	3.51×10^{-16}	2.68×10^{-13}	6.02×10^{-13}	3.46×10^{-13}	7.45×10^{-13}	3.47×10^{-13}	2.31×10^{-12}
Cm-248	1.12×10^{-16}	3.76×10^{-13}	8.45×10^{-13}	4.86×10^{-13}	1.05×10^{-12}	4.87×10^{-13}	3.24×10^{-12}
Co-60	3.72×10^{-3}	9.87×10^{-2}	1.20×10^{-1}	1.14×10^{-1}	2.34	6.25×10^{-1}	3.30
Cs-135	1.04×10^{-1}	1.15×10^{-1}	1.17×10^{-1}	6.96×10^{-2}	1.40×10^{-1}	6.12×10^{-2}	6.08×10^{-1}
Cs-137	7.86×10^{3}	7.41×10^{3}	6.14×10^{3}	3.53×10^{3}	7.59×10^{3}	3.54×10^{3}	3.61×10^{4}
Eu-150	7.35×10^{-7}	1.78×10^{-6}	3.01×10^{-6}	1.73×10^{-6}	3.72×10^{-6}	1.73×10^{-6}	1.27×10^{-5}
Eu-152	6.72×10^{-2}	1.74×10^{-1}	2.21×10^{-1}	1.30×10^{-1}	2.66×10^{-1}	1.23×10^{-1}	9.81×10^{-1}
Eu-154	4.19	1.05×10^{1}	7.14	6.54	2.18×10^{1}	8.12	5.83×10^{1}
Gd-152	1.86×10^{-13}	4.12×10^{-13}	6.77×10^{-13}	3.89×10^{-13}	8.37×10^{-13}	3.91×10^{-13}	2.89×10^{-12}
H-3	1.02×10^{1}	6.62	1.28	7.34×10^{-1}	1.58	7.37×10^{-1}	2.11×10^{1}
Ho-166m	3.99×10^{-6}	1.03×10^{-5}	1.79×10^{-5}	1.03×10^{-5}	2.21×10^{-5}	1.03×10^{-5}	7.49×10^{-5}
I-129	6.68×10^{-5}	5.51×10^{-5}	4.08×10^{-5}	2.33×10^{-5}	5.07×10^{-5}	2.35×10^{-5}	2.60×10^{-4}
In-115	8.81×10^{-12}	6.02×10^{-12}	1.75×10^{-12}	1.00×10^{-12}	2.16×10^{-12}	1.01×10^{-12}	2.08×10^{-11}
Kr-81[b]	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Kr-85[b]	N/A	N/A	N/A	N/A	N/A	N/A	N/A
La-138	1.55×10^{-10}	1.25×10^{-10}	7.45×10^{-11}	4.28×10^{-11}	9.21×10^{-11}	4.30×10^{-11}	5.33×10^{-10}
Nb-93m	1.11	1.30	1.44	8.25×10^{-1}	1.78	8.28×10^{-1}	7.27
Nb-94	1.29×10^{-5}	7.23×10^{-2}	1.62×10^{-1}	9.33×10^{-2}	2.01×10^{-1}	9.37×10^{-2}	6.22×10^{-1}
Nd-144	7.96×10^{-10}	7.06×10^{-10}	5.22×10^{-10}	3.00×10^{-10}	6.46×10^{-10}	3.01×10^{-10}	3.27×10^{-9}
Ni-59	0.00	8.98×10^{-2}	2.02×10^{-1}	1.16×10^{-1}	2.49×10^{-1}	1.16×10^{-1}	7.73×10^{-1}
Ni-63	0.00	4.80	9.29	5.96	1.05×10^{1}	3.34	3.39×10^{1}
Np-236	8.68×10^{-8}	1.32×10^{-6}	2.86×10^{-6}	1.64×10^{-6}	3.54×10^{-6}	1.65×10^{-6}	1.11×10^{-5}
Np-237	1.06×10^{-2}	7.75×10^{-3}	2.67×10^{-2}	6.43×10^{-2}	1.24×10^{-1}	3.14×10^{-2}	2.64×10^{-1}
Pa-231	1.42×10^{-4}	8.79×10^{-5}	7.44×10^{-6}	4.28×10^{-6}	9.20×10^{-6}	4.29×10^{-6}	2.55×10^{-4}
Pb-210	1.96×10^{-5}	1.17×10^{-5}	4.42×10^{-8}	2.54×10^{-8}	5.46×10^{-8}	2.55×10^{-8}	3.14×10^{-5}
Pd-107	8.90×10^{-3}	7.43×10^{-3}	4.80×10^{-3}	2.76×10^{-3}	5.93×10^{-3}	2.77×10^{-3}	3.26×10^{-2}

Table 3-1: DOE-estimated Radionuclide Inventory (Ci) decayed to January 1, 2016, Based on Assumed 5.1cm [2 in] Residual Calcine Depth Left After Retrieval (cont'd)

Radionuclide	CSSF 1	CSSF 2	CSSF3	CSSF 4	CSSF 5	CSSF 6	Total
Pm-146	2.89×10^{-4}	1.43×10^{-3}	2.84×10^{-3}	1.63×10^{-3}	3.51×10^{-3}	1.64×10^{-3}	1.13×10^{-2}
Pu-238	3.07	3.57×10^{1}	5.56×10^{1}	5.43×10^{1}	1.06×10^{2}	2.85×10^{1}	2.84×10^{2}
Pu-239	4.15×10^{-1}	8.01×10^{-1}	1.49	1.52	2.95	1.91	9.08
Pu-240	1.66×10^{-1}	6.33×10^{-1}	1.07	9.76×10^{-1}	2.08	1.03	5.96
Pu-241	1.15	1.79×10^{1}	2.85×10^{1}	2.61×10^{1}	5.64×10^{1}	2.80×10^{1}	1.58×10^{2}
Pu-242	9.58×10^{-5}	1.48×10^{-3}	2.73×10^{-3}	2.73×10^{-3}	4.88×10^{-3}	2.15×10^{-3}	1.41×10^{-2}
Pu-244	1.87×10^{-12}	2.32×10^{-12}	2.72×10^{-12}	1.56×10^{-12}	3.36×10^{-12}	1.57×10^{-12}	1.34×10^{-11}
Ra-226	4.25×10^{-5}	2.54×10^{-5}	1.75×10^{-7}	1.01×10^{-7}	2.17×10^{-7}	1.01×10^{-7}	6.85×10^{-5}
Ra-228	4.03×10^{-10}	2.90×10^{-10}	1.12×10^{-10}	6.45×10^{-11}	1.39×10^{-10}	6.47×10^{-11}	1.07×10^{-09}
Rb-87	1.58×10^{-5}	1.33×10^{-5}	8.82×10^{-6}	5.07×10^{-6}	1.09×10^{-5}	5.09×10^{-6}	5.90×10^{-5}
Se-79	2.65×10^{-2}	2.21×10^{-2}	1.67×10^{-2}	9.56×10^{-3}	2.07×10^{-2}	9.53×10^{-3}	1.05×10^{-1}
Sm-146	9.67×10^{-9}	2.87×10^{-8}	5.16×10^{-8}	2.97×10^{-8}	6.38×10^{-8}	2.98×10^{-8}	2.13×10^{-7}
Sm-147	5.64×10^{-6}	4.26×10^{-6}	2.03×10^{-6}	1.17×10^{-6}	2.51×10^{-6}	1.17×10^{-6}	1.68×10^{-5}
Sm-148	5.97×10^{-12}	1.05×10^{-11}	1.57×10^{-11}	9.03×10^{-12}	1.94×10^{-11}	9.06×10^{-12}	6.98×10^{-11}
Sm-149[a]	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Sm-151	1.60×10^{2}	1.07×10^{2}	4.51×10^{1}	2.25×10^{1}	6.27×10^{1}	3.40×10^{1}	4.31×10^{2}
Sn-121m	2.20×10^{-2}	5.06×10^{-1}	1.11	6.36×10^{-1}	1.37	6.38×10^{-1}	4.28
Sn-126	1.07×10^{-1}	8.90×10^{-2}	6.72×10^{-2}	3.85×10^{-2}	8.34×10^{-2}	3.87×10^{-2}	4.24×10^{-1}
Sr-90	6.53×10^{3}	6.53×10^{3}	5.26×10^{3}	3.27×10^{3}	7.03×10^{3}	3.06×10^{3}	3.17×10^{4}
Tc-98	3.79×10^{-7}	6.15×10^{-7}	8.75×10^{-7}	5.03×10^{-7}	1.08×10^{-6}	5.05×10^{-7}	3.96×10^{-6}
Tc-99	4.13	3.38	2.47	1.41	3.08	1.43	1.59×10^{1}
Te-123	1.16×10^{-14}	3.63×10^{-13}	8.00×10^{-13}	4.60×10^{-13}	9.89×10^{-13}	4.61×10^{-13}	3.08×10^{-12}
Th-229	9.23×10^{-8}	8.28×10^{-8}	6.26×10^{-8}	3.60×10^{-8}	7.74×10^{-8}	3.61×10^{-8}	3.87×10^{-7}
Th-230	9.82×10^{-4}	5.39×10^{-4}	2.05×10^{-5}	3.86×10^{-6}	2.38×10^{-4}	1.65×10^{-4}	1.95×10^{-3}
Th-232	4.06×10^{-10}	2.92×10^{-10}	1.15×10^{-10}	6.58×10^{-11}	1.42×10^{-10}	6.61×10^{-11}	1.09×10^{-9}
U-232	7.79×10^{-7}	3.88×10^{-5}	3.26×10^{-4}	2.28×10^{-4}	3.20×10^{-4}	7.70×10^{-5}	9.91×10^{-4}
U-233	1.53×10^{-6}	1.04×10^{-6}	4.22×10^{-6}	1.03×10^{-5}	1.77×10^{-5}	4.17×10^{-6}	3.89×10^{-5}
U-234	2.88×10^{-2}	2.94×10^{-2}	6.58×10^{-3}	5.95×10^{-3}	2.20×10^{-2}	1.69×10^{-2}	1.10×10^{-1}
U-235	2.00×10^{-4}	1.76×10^{-4}	6.49×10^{-5}	5.28×10^{-5}	2.84×10^{-4}	4.17×10^{-4}	1.19×10^{-3}
U-236	4.65×10^{-4}	4.45×10^{-4}	1.72×10^{-4}	1.44×10^{-4}	8.78×10^{-4}	9.30×10^{-4}	3.03×10^{-3}
U-238	1.13×10^{-5}	9.98×10^{-6}	1.14×10^{-5}	2.85×10^{-5}	1.55×10^{-4}	3.28×10^{-4}	5.45×10^{-4}
Zr-93	1.24	1.50	1.70	9.79×10^{-1}	2.11	9.82×10^{-1}	8.51

a. Stable isotope retained in table, but the inventory is listed as zero.
b. Noble gas retained in table, but the inventory is listed as not available.
Calcined Solids Storage Facility (CSSF)
not available (N/A)
1 Ci = 3.7×10^{10} Bq

One challenge of assessing CSSF waste inventory is that each bin contains multiple layers of chemically and radiologically different calcine (see Figure 3-1). Because of this heterogenous mixture, collecting and analyzing samples has been difficult, historically. Arguments are presented in engineering design file (EDF)-11126 (DOE-ID, 2021) regarding difficulty in representative sampling and difficulty mixing the sample. DOE states that calculating projected inventory using the HPM provides a verified alternative method. Additionally, the waste

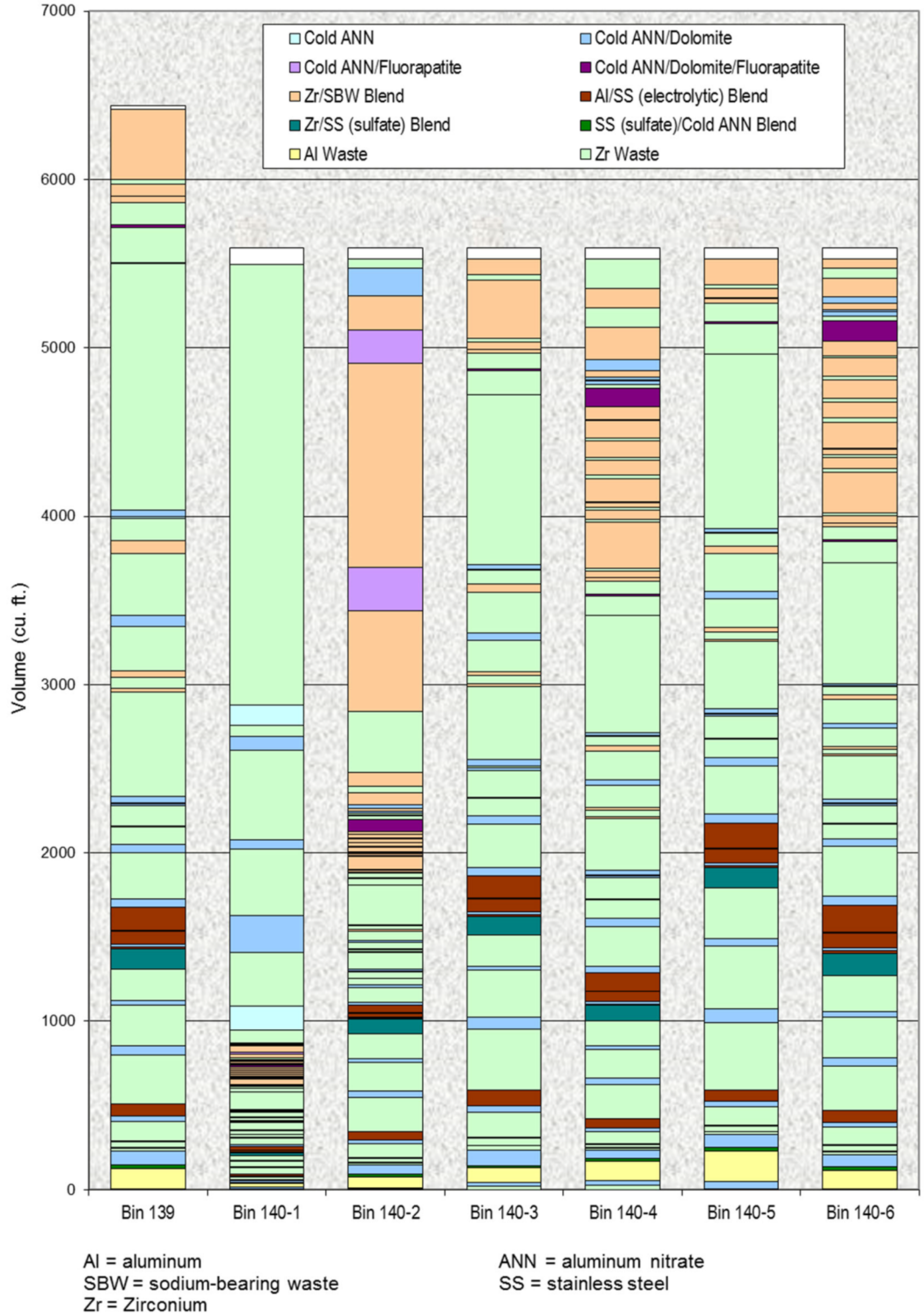


Figure 3-1: Chemically Different Calcine Layers in CSSF 3

composition is different for each of the six bin sets. It is important to note that calcine sample analyses are not available for all wastes that were calcined.

Additionally, DOE plans to transfer the waste from CSSF 1 to CSSF 6, which will change the volume and composition of waste in CSSF 6. As a result, the source term may need to be updated. DOE plans to leverage the pneumatic retrieval system's unique ability to access calcine at any depth to selectively remove calcine from the top or bottom of the bins during transfer and retrieval operations. This flexibility enables engineered retrieval strategies to minimize residual waste and radiation dose after closure beyond what is assumed in the waste inventory calculations and the PA [EDF-11126; DOE-ID, 2021)].

The initial screening approach described in WD Section 2.11.3.1 (described in Section 4.2.4 of this TER) narrowed the list of radionuclides to develop an initial inventory for use in the PA modeling. This approach resulted in an initial list of 75 radionuclides in each CSSF bin. DOE based the concentrations and inventory in the waste zone on ORIGEN2 modeling and scaling factors of hard to detect radionuclides to Cs-137. A limited number of liquid waste samples prior to calcination are used to validate the modeled values. DOE reduced inventories of volatile constituents such Np-237, I-129, and C-14 based on the expected volatilization of these constituents during the high-temperature calcination process. However, DOE has not reduced inventories of Tc-99 or Ru-106 because historical data are insufficient to determine accurate volatility adjustment factors (Staiger and Swenson, 2021). This results in additional conservativism for these radionuclides. NRC staff performed independent calculations of risk-significant radionuclides and verified the inventory calculations performed by DOE.

It is important to note that the HPM was developed in the 1990s and assumed the earliest retrieval date would be 2016. Therefore, the inventory calculations are based on decay until 2016, providing additional conservativism for future retrieval of calcine more than 10 years after the inventory calculated date. DOE states that the ingrowth of progeny between the assumed 2016 closure date and the future closure date would not significantly impact the dose results from the PA. NRC staff agree with DOE's statement that this decay date provides an upper bound for radionuclides that will continue to decay and is reasonable for ingrowth of progeny radionuclides.

Through various testing and mockups of the retrieval technology, DOE has determined that the height of residual calcine left in each bin after retrieval will be 5.1 cm [2 in] or lower. DOE assumes that the residual waste is a well-mixed fraction of all the waste that went into the SS storage bins. The inventory calculations are based on this assumption and would increase linearly with remaining calcine height. The resultant 5.1 cm [2 in] of waste remaining in the calcine bin sets correlates to more than 99 percent removal by volume. To determine waste inventory at closure, DOE scaled the waste inventory calculated based on decay to 2016 based on the percent volume of calcine remaining. Table 2-10 of the WD presents the assumed residual calcine volumetric percentages, which range from 0.33 percent to 0.97 percent across CSSF 1-6.

DOE considered inventory uncertainty by increasing the inventory by a factor of 5 in a one-factor at a time (OFAT) analysis. However, the waste zone was increased to 0.6 m [2 ft] instead of 0.3 m [1 ft], partially offsetting the impact of a higher inventory by diluting that inventory by a factor of 2. The inventory of Tc-99, Se-79, and Np-237 were varied by a factor of 2 in the probabilistic sensitivity and uncertainty analysis based on Staiger and Swenson (2021). Section 4.2.8 of this TER for more information on DOE's evaluation of inventory uncertainty. Additionally, NRC discussion and evaluation of the transfer line inventory and associated uncertainty is presented in Section 4.3 of this TER.

DOE does not provide an analysis demonstrating which HRRs would likely be remaining closure and in what concentrations, specifically. Rather, the WD includes a method of reducing the inventory, in aggregate, by more than 99 percent. As a result, NRC evaluation relies on the assumption that 5.1 cm [2 in] the calcine will remain in each bin set after bulk removal. NRC performed independent analysis of the waste inventory assuming only 99 percent of the calcine is removed from each bin set. Because the inventory would increase linearly with additional residual calcine, it is imperative that DOE demonstrate that only 5.1 cm [2 in] of residual calcine remains or adjust the radionuclide inventory and resultant PA based on actual residual calcine data. For example, assuming only 99 percent removal of each bin set may increase total inventory concentrations by more than 275 percent.

Additionally, NRC evaluation relied on the assumption that residual waste will be representative of the calcine waste before removal. NRC evaluation of DOE's assumption that the residual calcine will be a well-mixed fraction of all the waste streams in a specific CSSF bin is discussed in Sections 3.4.1 of this TER. NRC recommends that DOE consider sampling of the residual waste after waste retrieval to provide a better estimate of the remaining CSSF inventory following closure.

3.2 Identification of Highly Radioactive Radionuclides

HRRs are those radionuclides that contribute most significantly to risk to members of the public, workers, and the environment. In the context of the NRC staff's reviews of DOE's basis documents for WDs conducted under the NDAA, the term is not limited to radionuclides with high specific activity. The NRC staff considers the term "highly radioactive radionuclides," as used in the context of the NDAA, to be equivalent to the term "key radionuclides" used in the manual for DOE Order 435.1 (DOE M 435.1-1, "Radioactive Waste Management Manual"), and in some of the NRC staff's reviews of DOE basis documents for WDs. For radionuclides with initial insignificant inventories, the parents are included for consideration in the list of HRRs, as opposed to the specific progeny. Even though the specific progeny may not be listed as an HRR, it is still considered to be a key radionuclide.

Beginning with the radionuclide inventory described in Section 3.1 above, DOE developed the list of HRRs by identifying the radionuclides that were important in meeting performance objectives in 10 CFR Part 61, Subpart C, because they contribute to the dose to the workers, the public, and/or the inadvertent intruder in the CSSF PA. Table 3-2 presents the list of HRRs identified.

The list of HRRs includes all radionuclides listed in 10 CFR 61.55 Tables 1 and 2 that are present in the calcine. DOE considered performance assessment results from the all-pathways dose analysis and two inadvertent intruder analyses. NRC evaluation of the performance assessment is discussed in Section 4.3 of this TER. DOE indicates that the list of HRRs includes both short-lived radionuclides that may present risk without shielding and controls of direct exposure simply due to proximity to the HRRs and long-lived radionuclides that persist and may be mobile in the environment, potentially presenting risk of inhalation or ingestion to a member of the public.

The DOE analysis considered the projected inventories of these radionuclides at the time of site closure. Radionuclides with initial insignificant inventories were removed from consideration and their parents were included for consideration as HRRs. DOE examined resulting doses from the groundwater all-pathway analysis from the CSSF PA and OFAT cases to assess sensitivity and uncertainty for a 500,000-year post-closure period.

Table 3-2: Calcined Solids Storage Facility Highly Radioactive Radionuclides. Modified from Table 5-10 in DOE (2023).

Radionuclide	Performance Assessment Pathway[a]	10 CFR 61.55 Table
Am-241	Inadvertent intruder	Table 1
C-14	Insignificant to dose from air	Table 1
Co-60	Screened from further analysis in the CSSF PA/CA	Table 2
Cs-137	Insignificant to dose from inadvertent intruder	Table 2
H-3	Insignificant to dose from air	Table 2
I-129	Insignificant to dose from air	Table 1
Ni-63	Screened from further analysis in the CSSF PA/CA	Table 2
Np-237	Insignificant to dose from groundwater and inadvertent intruder	Table 1
Pu-238	Insignificant to dose for inadvertent intruder	Table 1
Pu-239	Insignificant to dose from groundwater and inadvertent intruder	Table 1
Pu-240	Insignificant to dose from groundwater and inadvertent intruder	Table 1
Pu-241	Insignificant to dose for inadvertent intruder	Table 1
Pu-242	Insignificant to dose from groundwater	Table 1
Sr-90	Insignificant to dose for inadvertent intruder	Table 2
Tc-99	Identified as an HRR from groundwater; insignificant dose for inadvertent intruder	Table 1
Alpha-emitting transuranic nuclides with half-life >5 yr[b]	Radionuclides not listed above were screened from further analysis in the CSSF PA/CA and are listed in Footnote b.	Table 1

a. Tc-99 is the only radionuclide that potentially may be a significant contributor to the very low doses to a member of the public or the hypothetical human intruder. All other HRRs identified under this column are identified as HRRs from Table 1 or 2 of 10 CFR 61.55.
b. Additional alpha emitting nuclides not already identified in the PA analysis as HRRs that have a half-life greater than 5 years were also included: Am-242m, Am-243, Cf-249, Cf-250, Cf-251, Cm-243, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, and Pu-244.

Additionally, DOE included HRRs identified in 10 CFR 61.55 that contributed potentially significant doses from the air all-pathway analysis for a 1,000-year post-closure period. DOE states that only Tc-99 may be risk-significant and most of the radionuclides included in the list of HRRs are not anticipated to be significant contributors to a potential future dose to a member of the public.[1]

Because DOE plans to remove the calcine in bulk, the mixed waste stream will be agitated and removed in aggregate, rather than separated by radionuclide, NRC has scoped its review of HRRs to further evaluate the retrieval testing data and assumptions of the retrieval technology.

[1] Am-241 was also found to be potentially important to the inadvertent intruder analysis (10 CFR 61.42).

For the purpose of identifying HRR to comply with Criterion (2) of NDAA Section 3116(a), NRC has evaluated the radionuclides selected and agrees that DOE has identified HRRs present in the waste, necessary for waste classification, and important to satisfying the performance objectives in 10 CFR Part 61, Subpart C. The definition of “HRRs” used by DOE appears to be consistent with the NRC staff’s understanding of the term as presented in NUREG-1854 (NRC, 2007). Specifically, the NRC staff agrees with DOE that HRRs are those radionuclides that contribute most significantly to radiological risk to the members of the public, workers, and the environment.

As DOE continues to evaluate assumptions in the CSSF PA as a result of consultation and monitoring activities, DOE should concurrently re-evaluate its list of HRRs as new information that could significantly change the results of its HRR evaluation becomes available.

3.3 Waste Retrieval Technologies

DOE has evaluated several different retrieval technologies and has defined (1) a bulk retrieval technology and (2) additional residual calcine removal technologies for cleaning of the bin sets. In developing various waste retrieval and transfer technologies, DOE noted the following parameters as significant inputs to their decision on a baseline technology:

- Calcine is a unique waste form, as it is a highly radioactive, dry, granular solid
- CSSF is not designed to receive, contain, or be decontaminated with liquid materials
- CSSF bins are mostly filled to capacity, limiting in situ treatment or fluid dissolution to remove calcine as a liquid slurry because there is not enough void space.
- DOE has evaluated numerous retrieval methods and has decided not to pursue development of such methods for the following reasons:
    - Sluicing or mixing would generate large volumes of waste that are less stable than calcine and would require additional storage, treatment, and disposal. Additionally, the CSSF bins do not have enough void space to add additional volume during the treatment and removal process.
    - Chemical cleaning: Adding acid to the calcine would create a gelatinous, semi-solid mass of partially dissolved material that would be more difficult to handle than its current dry granular form.
    - Mechanical manipulators: Mechanical equipment, such as an auger system, was eliminated because of the numerous moving parts that introduce reliability issues and difficulties with ensuring confinement of the calcine.
    - Robotic vehicles: The robotic technologies evaluated did not possess the physical flexibility to adequately clean all portions (e.g., ceiling, upper stiffening ring, walls) of the bins. Certain robotic technologies were chosen for further development to potentially remove residual calcine to meet the 5.1 cm [2 in] residual calcine depth criterion, which are discussed in Section 3.3.2 of this TER.

NRC has evaluated the additional technologies presented by DOE and corresponding justifications for not pursuing additional development and deployment. NRC agrees that, at this time, DOE has adequately considered residual cleanout technologies that account for the unique aspects of calcine waste. NRC understands that mockup testing results from the bulk retrieval method indicate that DOE may be able to remove calcine to the maximum extent practical without additional technologies. NRC agrees that DOE has performed sufficient qualitative evaluation of costs and benefits of the proposed retrieval methods to exclude additional retrieval methods from further development. NRC staff recommends that if results from operating the bulk retrieval system to remove calcine from the bin sets cannot achieve removal consistent with the assumptions presented in the PA (DOE-ID, 2022), DOE should assess the practicality of developing additional residual calcine removal technologies to meet NDAA Criterion (2).

3.3.1 Pneumatic Bulk Retrieval Technology

Ultimately, the baseline technology DOE selected is a dry pneumatic vacuum retrieval and transfer system that will agitate and remove calcine in bulk. DOE has performed at least 10 retrieval studies since 1975 and has researched the vacuum extraction and pneumatic transfer technology more than any other retrieval method.

The bulk retrieval system uses a pipe-in-pipe vacuum and compressed air system to retrieve calcine from the bottom of a bin (Figure 3-2). The system allows the operator to manipulate fluidizing air tube up and down with a linear slide, which moves the nozzle that delivers fluidizing air to the bottom of the vacuum. The pipe-in-pipe design reduces the area for the vacuum portion of the vertical line, which increases the velocity and aids in transport of solids. This technology leverages calcine's physical properties as a dry, granular solid. Based on testing results, the current configuration of the system can retrieve up to 544 kg/hr [1,200 lbm/hr]. This flow rate occurred when the air pressure at the top of the air tube was 65 psi and the nozzle was protruding 5.1 cm [2 in] out of the bottom of the vacuum pipe. Steady retrieval rates occurred when the nozzle was 1.27 cm [0.5 in] above the end of the vacuum line. Having the nozzle inside the vacuum line agitated enough material to fluidize it around the vacuum opening, resulting in an average 181 kg/hr [400 lbm/hr] transfer rate. This setup also had an air-to-particle ratio such that the flow out of the retrieval system was in dilute phase without significant accumulation on the bottom of the horizontal vacuum line. The bottom-up retrieval system will deliver calcine from the bottom of the bin group mockup to the bottom-up retrieval stand at the top of the bin and into the transfer system (Figure 3-3).

In addition to the pipe-in-pipe vacuum system, DOE has designed and tested an air lance to push residual calcine off the internal surface. The air lance creates a circular wind to agitate material and direct it toward the pipe-in-pipe vacuum.

During an onsite observation in 2022 as part of NRC's monitoring responsibilities for the INTEC facility, NRC staff received a briefing from DOE personnel on the design and operation of the bulk retrieval system, followed by a walkthrough of the associated piping infrastructure and a full-scale mockup of CSSF 1 (NRC, 2022a). This engagement provided NRC staff with a clearer understanding of the system's physical configuration and operational scale, enhancing the technical insights previously obtained through review of design drawings and documentation.

Based on historical and full-scale mockup design and testing, DOE has identified that one of the main challenges with calcine retrieval and transfer is access to the bins through the access risers. It is important to note that CSSF 1 does not have access risers. Certain pre-retrieval

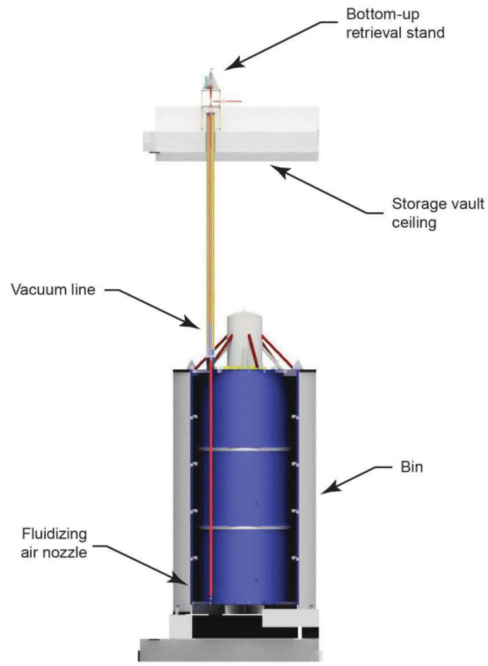


Figure 3-2: Bulk Retrieval System Installed on a Bin Set with Vacuum Line Extending Down to the Fluidizing Air Nozzle at the Bottom of the Bin (ICP, 2020)

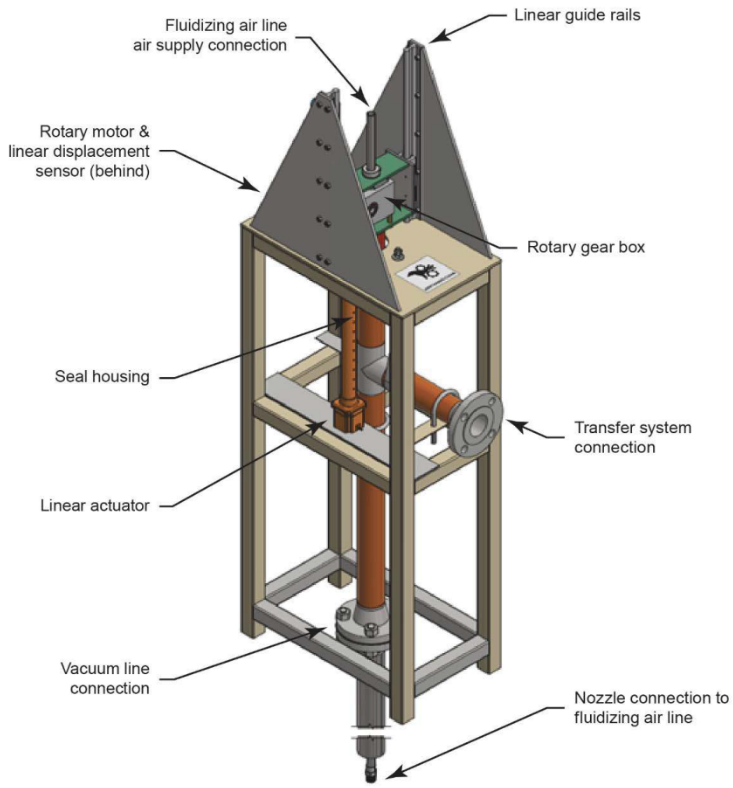


Figure 3-3: Bottom-up Retrieval Stand Assembly (ICP, 2020)

activities, supporting structures, and ancillary systems that establish access to the bins will be critical to ensuring that the retrieval of calcine remains safe for operators and is successful in meeting the 5.1 cm [2 in] residual calcine height. DOE has been developing techniques and technologies for vault mapping, vault coring, identifying access riser locations, access riser-vault interface, access riser positioning system, bin surface cleaning tool, access riser connection robot, and access riser cartridge (ICP, 2021). DOE has designed and tested certain technologies to ensure the pneumatic bulk retrieval system functions as designed, while other methods still require significant development.

In its RAI MEP-1, NRC staff stated that insufficient detail has been provided as to the specific iteration of the pneumatic bulk retrieval system that will be employed (NRC, 2024). DOE incorporated multiple references to the WD that presented results of historical and full-scale mockup testing of the current iteration of the bulk retrieval system. However, these reports cited multiple areas of future work and did not provide reasonable assurance that DOE has developed the appropriate technology, to date, to complete this work. In response to NRC's RAI MEP-1, DOE noted that retrieval tests at the full-scale mockup are ongoing and that additional tests, equipment improvements, and operational retrieval sequencing continue to be optimized based on previous retrieval tests. NRC staff understands that the bulk retrieval system that will be employed to transfer calcine from CSSF 1 to CSSF 6 has not yet been determined and that DOE will continue iterating the system to enhance performance.

NRC staff recognize that testing of the existing system has been successful in achieving the 5.1 cm [2 in] residual calcine simulant height assumed in the CSSF PA. Additionally, in Section 5.1.3 of the WD, DOE has committed to optimizing retrieval designs and operations to achieve removal of HRRs to the maximum extent practical. Evaluation of the full-scale mockup testing and DOE's ability to remove calcine to the maximum extent practical, as required by NDAA Section 3116 Criterion (2), is discussed further in Section 3.4.1 of this TER.

3.3.2 Additional Residual Calcine Removal Technologies

Based on DOE studies, a small amount of calcine may remain on the stiffening rings, floor, and other areas of the bins. Therefore, DOE has considered additional technologies to further remove remaining calcine using a residual cleanout system. These additional technologies will only be deployed if necessary to meet reduced residual waste volumes if the pneumatic bulk retrieval cannot achieve a residual calcine height of 5.1 cm [2 in]. One of DOE's main challenges is to develop a system that will introduce these tools through narrow 20-cm [8-in] diameter access risers and then remotely maneuver them within the bins. If additional removal of calcine is necessary, DOE has evaluated the following technologies:

- The snake-arm robot is a highly flexible robot specifically designed for working in confined and hazardous spaces. The electronic controls and motors are separate from the snake arm and thus are not affected by hazardous conditions, such as radiation. Though the snake-arm robot is proven technology that has been successfully used in various hazardous and complex environments, additional development and adaptation of the snake arm would be necessary to meet the needs of the Calcine Retrieval project as described in ICP (2017).

- The wall-climbing robot uses a vacuum system to adhere to interior walls of the bins, eliminating challenges of the self-supporting arm. Additional development is needed to optimize the vacuum system for reliability, develop and test compressed air or vacuum tooling for cleanout operations, and test strategies for deploying the crawler in the bin.

- The articulating continuum arm is long, flexible, cable-actuated retrieval system. One of the main challenges of this technology is that it may not be possible to directly reach the upper sections of the bin. If selected for additional removal of residual calcine, DOE will need to evaluate whether the arm will be able to perform the tasks required for residual cleanout.

In Section 5.1.3 of the WD, DOE has committed to continue to participate in technology exchanges and evaluate new retrieval technologies that may address known challenges or improve technologies or processes that have already been selected. NRC evaluation of the costs and benefits of additional removal of calcine are presented in Section 3.4 of this TER.

As noted in its response to RAI MEP-1, DOE is not considering the development of additional residual cleanout systems at this time due to the success of the current bottom-up retrieval unit and air lance systems in removing the calcine surrogate from the full-scale mockup. As previously mentioned, NRC staff recommends that if results from operating the bulk retrieval system to remove calcine from the bin sets cannot achieve waste retrieval consistent with the assumptions in DOE's CSSF PA (DOE-ID, 2022), DOE should assess the practicality of developing either a different residual cleanout method or additional residual calcine removal technologies to meet NDAA Criterion 2.

3.4 Removal to the Maximum Extent Practical

DOE has identified highly radioactive radionuclides and evaluated various retrieval methods to achieve removal the maximum extent practical, as required by NDAA Section 3116 Criterion 2. Based on DOE's current approach as described in the WD and supporting documentation, the pipe-in-pipe vacuum bulk retrieval system and associated air lance have been designed and tested. Specifically, DOE has developed a full-scale mockup of CSSF 1 and tested the bulk retrieval system using a calcine simulant, $CaCO_3$. However, DOE has not performed waste retrieval activities on any of the CSSF bins. Therefore, NRC has reviewed and evaluated the historical and current mockup testing design, procedures, and results. NRC staff understands that DOE will continue to optimize operation of the bulk retrieval system, evaluate options to overcome challenges to access the bins through the access risers, and consider potential alternative residual cleanout systems as necessary. This section presents DOE's testing results, NRC's evaluation of the testing results, and NRC's conclusion that highly radioactive radionuclides will be removed to the maximum extent practical.

3.4.1 Full Scale Mockup of CSSF 1

Although mock testing was not conducted specifically to demonstrate that removal to the maximum extent practical is feasible with the bulk retrieval system, NRC has assessed the results of full-scale mockup testing of CSSF 1 for demonstration of such removal. Of the six bin sets, CSSF 1 was chosen because the first phase of calcine retrieval consists of transferring approximately 220 m^3 [7,800 ft^3] of calcine from CSSF 1 to CSSF 6. Additionally, CSSF 1 has the most complex geometry, and DOE expects that retrieval from the other bin sets will be significantly easier due to the reduced number of internal obstructions, previously installed access risers, and simpler bin arrangement within the CSSF bin set.

CSSF 1 contains four composite Type-405 SS bin groups, a distributor pipe, a cyclone, and transport lines. Each composite bin consists of three concentric sub-bins. The innermost sub-bin (Bin A) in each group is cylindrical, has a diameter of approximately 0.9 m [3 ft], and is 7.6 m [25 ft] tall. One of the cylindrical sub-bins (Bin A), VES-WCS-115-4 in Figure 3-4 below, is approximately 8.5 m [28 ft] tall. Each cylindrical sub-bin is surrounded by an annular sub-bin (Bin B), which is, in turn, surrounded by a second annular sub-bin (Bin C). Small gaps between the sub-bins provide a path for airflow, which removes decay heat from the radioactive calcine. CSSF 1 bins contain internal obstructions such as thermowells, internally mounted wall stiffeners, bottom braces, and bin fill lines. Each of the cylindrical sub-bins (Bin A) has a centerline-mounted thermowell that extends from the top of the bin nearly to the bottom of the bin. Each of the annular sub-bins (Bins B and C) contains two main thermowells and at least one secondary thermowell. The two main thermowells are located near the center of the annulus (midway between the inside and outside walls), on opposite sides of the bin, and extend from the top of the bin nearly to the bottom of the bin. Each of the bins contains internal stiffening rings on the outer bin wall. The annular bins also contain internal stiffening rings on the inner bin wall as well as stiffeners on the flat bin floor. The bin floor stiffening ribs are 7.6 cm [3 in] tall. The stiffening rings on the outer and inner walls of the sub-bins are flat bars that extend 5.1 to 7.6 cm [2 to 3 in] from the walls. The outer wall stiffening rings in Bin C have a 5.1-cm [2-in] face that extends 2.5 cm [1 in] up and down from the ring.

Lastly, CSSF 1 does not have access risers, which presents an additional challenge for DOE to remove above grade structures, establish the access location, modify the facility, and develop systems to install the access risers for the retrieval system to enter the bins.

In its RAI MEP-1, NRC requested a comparison of full-scale mockup to actual CSSF Bins 2-6, including information on dimensions of bins, operation of pneumatic system that may differ as a result of different bin geometries, difference in total volume of material and any impacts that may incur, and estimated and/or volume of material remaining after retrieval operations (NRC, 2024). It is important to note that NRC received public comments related to the comparison of the bin sets, as well, because the information was lacking from the WD.

In the RAI response, DOE reiterated the bin geometry configurations and dimensions presented in the WD. DOE provided a short description of how the different heights on the bin sets may impact the ability for the bottom-up retrieval unit to remove calcine. It is important to note that the pipe-in-pipe retrieval system relies on the testing results that show the dry, granular calcine will be agitated and directed toward the bottom-up pipe-in-pipe vacuum. The bins in CSSF 1 range from 6.1 to 8.53 m [20 to 28 ft] in height, the shortest of any bins across all six CSSFs. Because the other bins are up to 240 percent taller than CSSF 1, DOE should account for the increased gravitational pressure and potentially decreased fluidization of calcine to the bottom-up retrieval unit for future testing and eventual retrieval operations. DOE provided clarity on how the difference in bin height may not be as significant a factor in retrieving calcine from greater depths:

> *Pneumatic transfer of calcine is less difficult to accomplish in a vertical transport line over large distances than in horizontal transport lines due to gravity affects in the horizontal pipes pulling the calcine to the bottom of the pipe. In the [bulk retrieval system] vertical transport lines, calcine remains near the center of the transport line, thus reducing the gravitational impact of calcine entrainment. Therefore, the impact of the difference in bin height is not considered a significant engineering factor in the retrieval of calcine from the bins.*

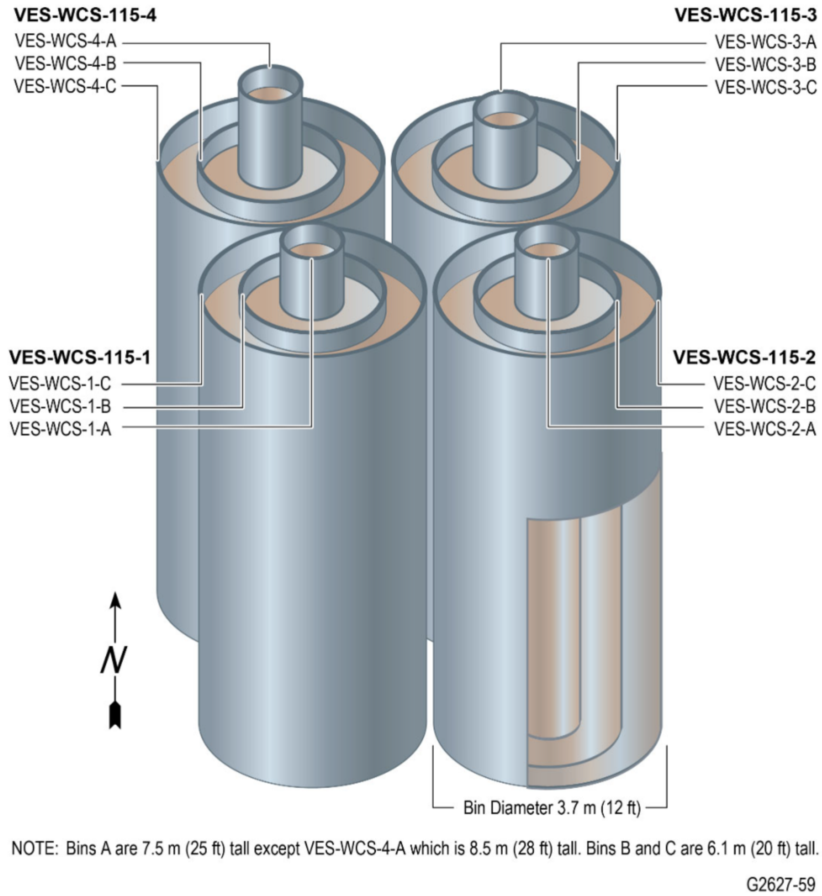


Figure 3-4: Calcined Solids Storage Facility 1 Nested Bin Configuration

An example of the height issue between bin sets is provided in Sandow (2021), which specifically calls out the height difference between the CSSF 1 mockup and the actual CSSF 6 cyclone height. The final system will have an additional elevation gain of approximately 12 m [40 ft] to get to the cyclone vault above the bins of CSSF 6. This vertical rise will occur at the side of CSSF 6 and will result in a higher-pressure loss due to the layout of the transfer piping than that of the integrated mockup. However, the system will need to be operated within the pressure boundaries based upon the pressure safety systems of CSSFs 1 and 6. This layout change is expected to result in a decrease in the material transfer rate in the final system compared to the integrated mockup.

While DOE does not consider bin height to be a significant factor that will affect the retrieval of calcine, DOE does present an additional engineering challenge related to difference in physical geometries of the bins: the location of the access riser with respect to the center of the bin. Because the vertical rise of CSSF 6 is offset from the center line, DOE anticipates increased pressure loss and must be able to compensate in some way to continue retrieval operations to the maximum extent practical. Additionally, NRC staff assume that calcine retrieval will be more difficult at greater depths. Because DOE has tested the bulk retrieval system using a full-scale mockup of CSSF 1, it is uncertain how DOE will compensate for the pressure loss due to the location of the transfer piping and additional height of CSSF 6 without additional testing or modeling.

DOE stated that to provide comprehensive testing of all aspects of the retrieval process, including installation, operation, maintenance and decommissioning, a full-scale mock-up of one bin in CSSF 6b is necessary, together with structures to represent the control room and rooftop levels of the building (AEA Technology, 2006). NRC staff agree that testing calcine simulant in a full-scale mockup of one bin from CSSF 6 would provide further data on operating parameters to test the bulk retrieval system in the bin with the greatest height, diameter, and total volume across the CSSF.

As compared to the geometry of CSSF 2 through 6, NRC agrees that CSSF 1's cylindrical and annular bin configuration as well as the multiple internal obstructions present the most challenging environment for physical manipulation of the bulk retrieval system. The full-scale mockup of CSSF 1 demonstrates a realistic physical geometry and presents additional opportunities for testing of the installation of access risers to the bin sets. When accounting for elevation change and additional gravitational impact of calcine retrieval at greater depths in other CSSF bin sets, DOE should continue to monitor pressure boundaries and adjust the piping system parameters and location within the bin to achieve sufficient agitation and suction when removing calcine.

Additionally, DOE's demonstration of its ability to remove HRRs to the maximum extent practical during operations relies on the assumption that the waste will be well mixed and removed in aggregate using a bottom-up retrieval system, rather than partitioning and selectively removing HRRs. As shown in Figure 3-1, the calcine in the bins is heterogeneous. Each bin in CSSF 3 contains multiple layers of chemically and radiologically different calcine, and DOE assumed that the other bin sets will also have multiple bin segments that may contain multiple layers of calcine. Limited data from liquid waste samples taken prior to calcination was used to develop the inventory for the PA calculations. Due to the number of assumptions made in estimating the radionuclide inventory of the residual waste (see Sections 3.1 and 4.2.4), there is significant uncertainty in the inventory of HRRs expected to remain following waste retrieval operations. Because of the importance of the final HRR inventory to (i) dose and (ii) demonstration of removal to the maximum extent practical based on risk arguments, NRC staff recommends that DOE provide additional support for the final inventory estimates and consider sampling the residual calcined waste following waste retrieval operations.

3.4.2 Calcine Simulant and Material Properties of Calcine that May Impact Retrieval Efficacy

The full-scale mockup testing of CSSF 1 has been tested on calcine simulant, $CaCO_3$. To date, DOE has not performed waste retrieval activities on any of the CSSF bins. Rather, DOE has used calcium carbonate simulant on multiple tests of the bulk pneumatic retrieval operations since 1979, aiming to simulate attributes of calcine such as bulk density, friction flow angle, and transport particle attrition. In 1981, a consulting firm specializing in bulk solids flow in bins and hoppers recommended a special grade of $CaCO_3$ as a material with flow properties very similar to the fines in zirconia-type calcine (Westra, 1982). In 1991, DOE studied the free-flowing properties of calcine, tested samples of the different types of calcine, and identified a suitable, nonhazardous simulant to support retrieval system design–a coarse limestone $CaCO_3$ -14 mesh +60 mesh, with a particle size between 250 and 1,410 microns [0.0098 to 0.056 in]. It is important to note that this simulant was selected with specific regard to bulk handling of calcine as opposed to retrieval and pneumatic transport (Wanke, 1993). In 2005, testing used both untreated, inactive calcine produced during the commissioning of the New Waste Calcining Facility (NWCF) at INL and caked calcine simulant produced by the addition of water to the calcine. This calcine simulant, known as T2 calcine, is non-radioactive calcine which was formed in the NWCF calciner. While T2 particles are spherical and have a bulk density similar to the CSSF 1 calcine, T2 particle size is smaller than that of calcine. DOE concluded that T2 calcine would be the most representative simulant to use for retrieval and transport testing (Lower, 2016). However, there is a limited supply of T2 calcine, which is being reserved for future treatment studies. Therefore, DOE states that calcium carbonate should be used as a simulant for CSSF 1 alumina calcine in a particle size range that most closely resembles the particle size distribution of the sampled alumina calcine from CSSF 2, which is the most representative of calcine stored in CSSF 1 (AEA Technology, 2006). DOE concluded that calcium carbonate is a conservative simulant with regards to alumina calcine because calcium carbonate is heavier than calcine with rough particles that are angular in shape.

In 2016, DOE reported the physical and thermal properties for CSSF 1 calcine in a referenceable compilation of material property data (Lower, 2016). Although a detailed breakdown of particle size distribution does not exist, DOE stated that calcine consists of a bimodal particle distribution of product and fines. The calcine contains large amounts of calcine product mass mean particle diameter (MMPD) ranging between 550 and 600 microns [0.0217 and 0.0236 in] for CSSF 1 calcine} and a large amount of fines {MMPD estimated to be approximately 10 microns [0.0004 in]} with very little material in the ranges between. Historical samples from the top, middle, and bottom of CSSF 2 bin 3 have shown that calcine follows the typical particle distribution for zirconia calcine. The MMPD for alumina calcine stored in CSSF 2 bin 3 ranged between 560 and 700 microns [0.0220 and 0.0276 in], while the MMPD for calcine product stored CSSF 1 ranged between 550 and 600 microns [0.0217 and 0.0236 in]. Since particle size distribution data for CSSF 1 are not available, it is believed that the retrieved samples from CSSF 2 provide the most accurate representation of the particle size distribution expected in CSSF 1. It is important to note that DOE estimates almost 30 percent reduction in calcine particle size due to particle attrition after a transported length of 90 m [295 ft].

NRC staff requested additional information regarding expectations with respect to the retrievability of waste. In AEA Technology (2006), DOE described calcine as potentially cohesive and hardened and notes that during retrieval technology testing that there was some success breaking up and clearing hardened calcine. DOE reports indicate that it was possible to block the nozzle, and some material proved to be “intractable.” NRC staff were unsure if hardened waste was expected to be present or if it was assumed to be present to help optimize waste retrieval if waste was more recalcitrant to retrieve than expected.

In its response, DOE noted that historical testing has indicated that an easily penetrable crust had formed in the top of the Zr calcine, but that none was encountered in the aluminum (Al) calcine (Staples et al., 1979) and it does not expect to encounter hardened calcine during actual retrieval operations. Operating parameters and calcine storage features, such as temperature and precipitation/moisture run-in controls, prevent calcine agglomeration, which would hinder future calcine retrieval. Peak temperatures from radiolytic decay have not exceeded caking temperatures for the aluminum calcine, zirconium calcine, or sodium-bearing waste. Additionally, DOE confirmed that significant agglomeration has not occurred in the distributor pipe or at the top of the bins in CSSF 1 during an enhanced video inspection in 2016 (Coughlan, 2024).

DOE has not performed visual inspection or retrieved samples from the middle or bottom of the bin sets due to practical challenges, worker dose considerations, and additional waste generation concerns. NRC staff agree with DOE's statement that "the amount of material left over is highly dependent upon material properties including bulk density, particle size, and angle of repose," particularly because the calcine waste is stored in radiologically and chemically different layers (Sandow, 2021). The results of the dose assessment presented in the CSSF PA are highly dependent on the distribution and total amount of residual calcine remaining in the bin sets. NRC staff recommends that DOE evaluate additional radiological and chemical properties of calcine relevant to projections of residual calcine after waste retrieval activities commence to optimize waste retrieval, including sampling following termination of waste retrieval operations to support final inventory estimates. DOE should provide updated PA calculations in case radionuclide concentrations in residual calcine exceed projections of prior assessments.

3.4.3 Criteria for Termination of Retrieval Operations

In Section 5.1.1 of the WD, DOE states that testing of the full-scale mockup of CSSF 1 was critical to establish criteria for ending calcine retrieval activities (i.e., determine when the maximum extent practical has been achieved). However, neither the WD nor supporting documentation discuss the criteria DOE plans to use to determine when removal to the maximum extent practical has been achieved. Therefore, NRC assumed that previous testing may be used as a benchmark to determine when to cease retrieval operations but did not assume that mock testing is intended to demonstrate removal to the maximum extent practical. Based on historical testing, different studies have used different criteria to determine when to cease operations. Reports from testing from fall 2005 note that 99 percent of calcine simulant was removed from a depth of 2.4 m [8 ft] from a smaller scale mockup of the basic geometry of a bin in CSSF 6. Based on visual assessment, the mean thickness of residual calcine is less than 2.54 cm [1 in] across the bottom of the tank (AEA Technology, 2006). However, reports from testing in 2020 and 2021 state that retrieval activities "with the manual air lance continued until minimal retrieval rates were occurring" during Bin A Bulk Retrieval Testing but does not specifically cite what those rates were that triggered cessation of operations (Sandow, 2021). From these two tests, it appears that different criteria were used to assess when to terminate operations. However, Section 5.1.3 Footnote 57 of the WD acknowledges that 99 percent removal of HRRs within a particular bin is not, by itself, a justification for stopping HRR removal activities. For comparison, the PA assumes the volume of residual waste is 5 cm [2 in], which is the basis for the long-term dose estimates.

In its response to NRC RAI MEP-2 on this topic, DOE stated that, to date, retrieval testing has not been conducted to establish the criteria to determine when waste retrieval has proceeded to the maximum extent practical and retrieval operations can be terminated. Furthermore, DOE stated:

> *The current plan for evaluating when to stop retrieval will consist of a combination of video monitoring, observations of the changing depths and volumes of calcine during retrieval, and the point of diminishing returns (i.e., monitored retrieval rates) based on continued air lance movement of calcine and attempts to remove the calcine with the bottom-up retrieval unit. These criteria will be developed after designs are finalized, integrated testing completed, and operating parameters established. The retrieval equipment designs, and operating parameters continue to be optimized with ongoing retrieval testing.*

DOE also stated that, as with all retrieval systems, a point of diminishing returns will occur in which continued operation of the system does not result in additional removal of calcine in terms of depths or volumes, such that continued operations are not reasonable. NRC evaluation of the costs and benefits of additional removal is presented in Section 3.4.4 of this TER.

Moreover, NRC staff requested additional information on how DOE plans to verify that retrieval operations have been successful in removing HRRs to the maximum extent practical during and after operations. DOE stated that it will use video monitoring and monitor the changing depths and volumes of calcine during operations. NRC staff recommend DOE incorporate monitoring the retrieval rates of calcine through the pipe-in-pipe vacuum, as well. While increasing or decreasing retrieval rates may be indicative of other physical phenomenon (e.g., nozzle is blocked, calcine has agglomerated, pipe connection and/or seal has been compromised, etc.), DOE will be able to best monitor the efficacy of calcine removal by monitoring retrieval rates in tandem with video monitoring and changing depths and volumes of residual waste.

NRC staff understand that the final iteration of the bulk retrieval system has not yet been developed, as noted in Section 3.3.1 of this TER. However, to determine when removal to the maximum extent practical has been achieved, it is imperative that DOE develop criteria for termination of retrieval operations and develop plans to verify that the criteria have been met. Results from testing the retrieval of calcine simulant from a full-scale mockup of CSSF 1 have demonstrated that the bulk retrieval system is able to remove more than 99 percent of calcine simulant by volume. A key assumption in the CSSF PA is that 5.1 cm [2 in] of residual calcine is distributed evenly across the floor of the bins. NRC staff recommends that DOE incorporate this assumption into its criteria for termination of retrieval operations such that the final configuration of the bin sets align with the assumptions modeled in the CSSF PA (DOE-ID, 2022).

3.4.4 Costs and Benefits of Additional Removal of Waste

In demonstrating that removal to the maximum extent practical will be achieved, DOE performed a cost-benefit analysis to account for worker dose, technology development and deployment, and potential generation of additional waste as a result of retrieval and treatment operations. Section 5.2 of the WD presents a description of the analysis and references cost-benefit analyses performed since 2016. NRC has evaluated DOE’s assessment of costs and benefits of additional removal of calcine (and the HRRs therein) beyond what will be performed by the bulk retrieval system. This section describes DOE’s justification and NRC’s evaluation of developing the residual cleanout technologies discussed in Section 3.3.2 of this TER, namely the wall-climbing robot and the articulating continuum arm. Section 3.3 of this TER discusses the costs and justification of not pursuing certain calcine removal methods.

DOE states that further removal of residual waste beyond the 99 percent total volume reduction {5.1 cm [2 in] of residual waste calcine depth} “would be impractical, increase costs, add schedule delay, increase the potential risk to workers, and result in an insignificant reduction in the very low potential doses to the public and the hypothetical human intruder.” As a result, DOE does not plan to develop a residual waste cleanout system to remove additional waste, as discussed in Section 3.3.2 of this TER. DOE conducted a 5-year rough-order of magnitude estimate of approximately $52 million for the full-scale mockup, transfer of calcine from CSSF 1 to CSSF 6, and placement of CSSF 1 in a safe configuration. NRC staff understands that the full scope of work to retrieve calcine from each bin set will take significantly more resources, including time and funding.

Based on the CSSF PA results, the maximum effective dose for the acute inadvertent intruder scenario is 0.071 mSv/yr [7.1 mrem/yr] 500 years after closure, based on a residual calcine depth of 5.1 cm [2 in] in each bin set. The maximum effective dose for the chronic inadvertent intruder scenario is 0.036 mSv/yr [3.6 mrem/yr] 500 years after closure period. The acute and chronic inadvertent intruder doses are considerably less than the 5 mSv/yr [500 mrem/yr] dose NRC staff uses to evaluate the 10 CFR 61.42 performance objective. Lastly, DOE-ID calculates a maximum effective dose to a member of the public of 0.000345 mSv/yr [0.0345 mrem/yr] during a 10,000-year evaluation period, which is considerably less than the 0.25 mSv/yr [25 mrem/yr] performance objective.[2] Because these effective doses are significantly lower than the performance objectives, DOE stated that regardless of cost or efficiency, any incremental reduction of dose in any of these scenarios would not significantly reduce the risks.

NRC guidance document NUREG-1854 (NRC, 2007) states that costs could include, but not necessarily be limited to, financial costs, delays, increases in risks to workers and members of the public, system impacts (e.g., generation of secondary waste streams requiring storage in tanks), and transportation risks (if waste is moved offsite for disposal). Benefits may include, but not necessarily be limited to, decreases in radiological risks to workers and members of the public (including inadvertent intruders), reduction in impacts on natural resources, and reduction in costs of other entities incurred because of effects on natural resources. DOE provided financial costs of developing additional removal technologies to clean residual calcine from each of the 6 CSSF bin sets. DOE concluded that because of the high cost of developing these technologies compared to the already low effective doses indicated in the CSSF PA, DOE does not plan to develop any additional residual cleanout technologies. However, DOE did not provide quantitative analysis demonstrating an increase in risk to workers performing additional calcine removal or an analysis of system or environmental impacts. In any case, NRC agrees that if DOE performs retrieval operations and can verify that residual calcine meets the assumptions modeled in the CSSF PA, additional reduction in waste volume in the CSSF bin sets would not result in significant reductions in risk based on DOE’s dose estimates. In Section 5.1.3 of the WD, DOE has committed to continue to participate in technology exchanges and evaluate new retrieval technologies that may address known challenges or improve technologies or processes that have already been selected.

3.5 NRC Review and Conclusions

NRC has performed a detailed review and evaluation of the information submitted by DOE, including the WD, CSSF PA, and supporting references. NRC agrees that DOE will be able to achieve removal of highly radioactive radionuclides to the maximum extent practical using the bulk pneumatic retrieval technology that includes the pipe-in-pipe vacuum and air lance. These conclusions are based on the following assumptions:

- Residual calcine depth of 5.1 cm [2 in] in each of the bins, representing a more than 99 percent volume reduction.
- Radiological inventory lower than assumed in the PA (see Sections 3.1 and 4.2.4 of this TER).
- The bulk removal of calcine will provide sufficient mixing of residual calcine such that specific highly radioactive radionuclides are not left concentrated in the layer of residual waste remaining at the bottom of the bin sets.
- DOE will develop criteria to determine when removal to the maximum extent practical has been achieved to support subsequent termination of retrieval operations.

[2] The overall peak dose is 0.002 mSv/yr [0.2 mrem/yr] at 19,500 years.

- DOE should assess the practicality of developing additional residual cleanout technologies if the bulk retrieval system does not achieve removal to the maximum extent practical during operations.

NRC also makes the following recommendations to support DOE’s conclusions regarding Criterion 2:

- DOE should consider sampling the residual waste following waste retrieval operations to assess the validity of CSSF PA assumptions made in developing the final estimated inventory and adjust PA calculations if the inventory of HRRs is significantly higher than assumed in the PA.

- DOE should continue iterating its bulk retrieval system and optimize calcine retrieval to achieve removal to the maximum extent practical for bin sets with waste depths greater than that of the full-scale mockup for CSSF 1.

- DOE should sample residual calcine to verify its radiological and chemical properties and concurrently re-evaluate the validity of assumptions of the calcine simulant, optimize operating parameters of the bulk retrieval system, and update the CSSF PA inventory assumptions as new information becomes available that could significantly increase the dose estimates.

- DOE should consider a full-scale mockup of CSSF 6, as recommended in AEA Technology (2006).

4.0 CRITERIA THREE (A) AND THREE (B)

(3) (A) does not exceed concentration limits for Class C low-level waste as set out in Section 61.55 of Title 10, Code of Federal Regulations, and will be disposed of–

(i) in compliance with the performance objectives set out in Subpart C of Part 61 of Title 10, Code of Federal Regulations; and

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

(B) exceeds concentration limits for Class C low-level waste as set out in Section 61.55 of Title 10, Code of Federal Regulations, but will be disposed of–

(i) in compliance with the performance objectives set out in Subpart C of Part 61 of Title 10, Code of Federal Regulations;

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

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

Before DOE can determine whether Section 3116 (a)(3)(A) or (a)(3)(B) above applies, it must first determine whether the waste exceeds concentration limits for Class C LLW provided in 10 CFR 61.55. After applying the NRC's guidance on classification of waste that is incidental to reprocessing found in Section 3.5 of NUREG-1854 (NRC, 2007), DOE believes that the calcine waste at closure will not contain concentrations greater than the limits for Class C waste. Therefore, NDAA Section 3116(a)(3)(A) applies to calcine waste (see Section 4.1.1).

Whether the waste is greater than or less than Class C, DOE must also demonstrate that the waste will be disposed of in compliance with the performance objectives in Subpart C of 10 CFR Part 61 and pursuant to the state-approved closure plan as part of the Hazardous Waste Management Act/ Resource Conservation and Recovery Act (HWMA/RCRA) Partial Permit. The performance objectives require protection of the general population from releases of radioactivity, protection of individuals from inadvertent intrusion into the waste, protection of individuals during operations, and stability of the disposal site after closure. Protection of the general population (including inadvertent intruders) is typically demonstrated through a PA calculation that takes into account the relevant physical processes and the temporal evolution of the system.

4.1 Assessment of Waste Classification

Section 3116 of the NDAA lists two sets of criteria for non-HLW: Section 3116 (a)(3)(A) and (a)(3)(B). The applicable set of criteria is dependent on the classification of waste. If calcine waste is Class C (or less), then the criteria in (a)(3)(A) apply. If the waste is greater than Class C, then additional criteria provided in Section 3116 (a)(3)(B) apply.

LLW intended for near-surface disposal is classified as Class A, B, or C based on concentration limits provided in 10 CFR 61.55. The 10 CFR Part 61 waste classification system was designed to protect an inadvertent intruder. To this end, additional requirements are specified in 10 CFR Part 61 for Class B and C wastes, which pose a greater potential risk to an inadvertent intruder than Class A waste. To determine if calcine waste is Class C or less, DOE must compare concentrations of radionuclides in CSSF components (e.g., tanks and piping) to the limits provided in 10 CFR 61.55. The concentration limits in 10 CFR 61.55, Table 1, for long-lived radionuclides include limits for a general class of radionuclides (alpha-emitting transuranic radionuclides with half-lives greater than five years) that DOE must identify.

Table 4-1 presents DOE's waste classification results for (1) an example CSSF bin and (2) for transport lines, respectively. DOE calculated waste inventory for the bin sets based on a residual waste depth of 5.1 cm [2 in] that is evenly spread across the floor of the bins. For the transport lines, DOE assumed approximately four percent of the pipe volume contains waste that is spread evenly over the 20.3-cm [8-in] drill hole diameter, resulting in a waste thickness of 0.113 cm [0.044 in]. DOE did not include the CSSF 1 waste profile for the transport lines because these lines were closed during a previous HWMA/RCRA action. DOE concluded that the waste will not be greater than Class C at the time of closure.

DOE used an approach consistent with Category 3 of the NRC's concentration averaging guidance described in Section 3.5 and Appendix B of NUREG-1854. DOE developed site-specific averaging expressions for CSSF, based on the results of the CSSF PA inadvertent intruder analyses. These site-specific averaging expressions include a site-specific factor that accounts for the unique conditions at CSSF that differ from the assumptions inherent to the generic 10 CFR Part 61 waste classification limits (e.g., drilling scenario, thinner waste layer). NRC's waste classification tables listed in 10 CFR 61.55 are based on the basement excavation scenario, which represents exposure of an inadvertent intruder to waste in a commercial shallow land burial site with waste.

At the time of closure, the residual waste within the bins will be a minimum of 13.7 m [45 ft] below the ground surface and will be grouted to provide a barrier against intrusion. The residual waste within the transport lines will be approximately 3.5 m [11 ft] below the ground surface at its shallowest point, and waste transfer equipment that cannot be removed will be stabilized with grout within the bin or storage vault. Additionally, the transport lines are encased in reinforced concrete for shielding. The waste classification calculations were based on both acute and chronic exposure scenarios that were evaluated for each of the CSSF bins and transport lines.

DOE has excluded the basement excavation scenario because the bin set waste disposal depth will be greater than 4.9 m [16 ft] and grouted, which DOE considers as a robust barrier that precludes inadvertent intrusion directly into the residual waste until 500 years post-closure. Instead, DOE determined that the more appropriate and credible scenario for potential human intrusion is one that assumes that a hypothetical intruder inadvertently drills a well through a bin set after the assumed period of institutional control ends. DOE considered two types of exposure scenarios in the CSSF PA to estimate dose to the hypothetical intruder: acute and chronic exposure scenarios. Acute scenarios evaluated the dose received from drilling a well and subsequent exposure to residual waste in the drill cuttings; acute exposure is evaluated over a short time period. Chronic scenarios evaluated the dose received from spreading the drill cuttings over a specific area while living and/or working on that area. Table 4-2 presents detailed descriptions of these scenarios. DOE has employed the same approach to evaluating the waste classification of transport lines. NRC staff note that the transport line residual waste disposal depth will not be greater than 4.9 m [16 ft] but will be greater than 3.0 m [10 ft] at its shallowest point and will contain a robust barrier to protect against inadvertent intrusion.

Table 4-1: DOE Class C Sum of Fraction (SOF) Results for the Bin Sets and Transport Lines Based on Projected Intrusion in 2516

Bin Set or Equipment	Acute Class C Equation		Chronic Class C Equation	
	Table 1 SOF	Table 2 SOF	Table 1 SOF	Table 2 SOF
CSSF 1 Bin Set	2.1×10^{-2}	1.4×10^{-3}	6.2×10^{-3}	3.9×10^{-4}
CSSF 2 Bin Set	3.5×10^{-2}	7.8×10^{-4}	9.9×10^{-3}	2.2×10^{-4}
CSSF 2 Transport Line Waste Profile	7.8×10^{-4}	1.7×10^{-5}	2.2×10^{-4}	4.9×10^{-6}
CSSF 3 Bin Set	5.4×10^{-2}	6.3×10^{-4}	1.5×10^{-2}	1.8×10^{-4}
CSSF 3 Transport Line Waste Profile	1.2×10^{-3}	1.4×10^{-5}	3.4×10^{-4}	4.0×10^{-6}
CSSF 4 Bin Set	9.9×10^{-2}	9.2×10^{-4}	2.8×10^{-2}	2.6×10^{-4}
CSSF 4 Transport Line Waste Profile	2.2×10^{-3}	2.0×10^{-5}	6.2×10^{-4}	5.8×10^{-6}
CSSF 5 Bin Set	9.2×10^{-2}	9.5×10^{-4}	2.6×10^{-2}	2.7×10^{-4}
CSSF 5 Transport Line Waste Profile	2.1×10^{-3}	2.1×10^{-5}	5.8×10^{-4}	6.0×10^{-6}
CSSF 6 Bin Set	3.2×10^{-2}	3.4×10^{-4}	8.9×10^{-3}	9.6×10^{-5}
CSSF 6 Transport Line Waste Profile	7.0×10^{-4}	7.5×10^{-6}	2.0×10^{-4}	2.1×10^{-6}

Table 4-2: Summary of Exposure Scenarios for Inadvertent Intrusion into the Calcined Solids Storage Facility Bins and Transport Lines.

Exposure Scenario	Description
Acute Exposure: drilling	Assumed to occur any time after 500 years post-closure for the bins and transport lines. Exposure to the residual radioactive waste is assumed to occur as a result of drilling an agricultural (i.e., large diameter) well through the bins or transport lines. Evaluated exposure pathways include external, inhalation, and soil ingestion for a time period required to complete the well.
Chronic exposure: post-drilling agriculture	Assumed to occur any time after 500 years post-closure for the bins and transport lines. Exposure to the residual radioactive waste is assumed to occur as a result of drilling a residential water supply well through the bins or transport lines, mixing exhumed drill cuttings and waste with garden soil, and using the soil for growing crops and beef. Assumed exposure pathways include direct exposure to contaminated soil, inhalation of contaminated soil, ingestion of contaminated garden soil, ingestion of vegetables grown in contaminated garden soil, and ingestion of contaminated beef and milk.

DOE’s approach is a departure from the guidance presented in NUREG-1854 Section 3.5.1.1, which provides examples of where site-specific averaging for waste classification should apply. Because the majority of the transport lines are buried less than 4.9 m [16 ft] deep, NUREG-1854 indicates that they should be evaluated according to the “Shallow waste, intruder barrier” scenario, which correlates to a basement excavation scenario for the acute construction worker or chronic resident receptor type at 500 years after closure. However, DOE has evaluated residual waste in both the bin sets and transport lines using the acute and chronic well driller receptor types. NRC conferred on this approach to assessing waste classifications for the tanks in the TER for the Draft WIR Evaluation for Closure of Hanford Site WMA C (NRC, 2020). NRC determined use of the well driller scenario was acceptable for the transport lines at the INL CSSF because the waste is buried at least 3 m [10 ft] below the ground surface and the volume of the waste in the pipe is relatively small compared to the basement excavation scenarios, resulting in significant dilution in the excavation scenario. NRC has evaluated the approach used by DOE to perform waste classification calculations for the CSSF and performed independent verification of DOE's waste classification calculation results. NRC staff note the following differences in the parameters for waste classification calculations between the CSSF and Hanford Site WMA C sites:

- CSSF assumes an exposure time for drilling that is four times greater than that of the Hanford facility, representing additional conservatism.
- CSSF assumes the residual waste within transport lines is spread evenly over the 20.3 cm [8 in] drill hole, which represents additional conservativism from the Hanford facility's assumption that the residual waste was spread evenly across the internal surface of the pipeline.
- CSSF assumes only 5.1 cm [2 in] of waste will remain in the bin sets, resulting in significantly less residual waste volume and mass than the Hanford facility's calculations.
- CSSF assumes drilling into transport lines could occur 500 years after closure, resulting in additional decay compared to the Hanford approach of assuming drilling into pipelines could occur 100 years after closure.
- CSSF intruder scenarios does not assume a closure cap is placed over the residual waste. However, DOE stated the CSSF will potentially be covered with an engineered cap, which will be designed at a later date, representing additional conservatism.
- CSSF adopted NRC's recommendation from the Hanford facility TER to exclude the factor of 4 from Equation 6-1 in the WD, which represents the averaging expression used to determine the individual radionuclide contribution to the sum of fractions (SOF) based on the acute drilling scenario. This factor is a volumetric dilution factor used by NRC in the basement excavation scenario for the Class C intruder analysis. NRC assumed the concentrations to which the acute intruder was exposed were not mixed with soil (unlike the chronic intrusion case); therefore, this factor was not applied to the acute intruder.

As discussed in NUREG-1854, the 10 CFR 61.55 limits were based on an intrusion dose of approximately[3] 5 mSv (500 mrem), with certain adjustments applied to account for waste accessibility to an inadvertent intruder, dilution with waste below the concentration limits, and radionuclide inaccessibility in the wasteform. The differences between the Class A and Class C

[3] The waste classification limits were based on individual organ dose limits using ICRP 2 methodology, at levels the NRC has determined is approximately equivalent to 5 mSv (500 mrem) total effective dose equivalent (TEDE).

limits include 400 years of additional decay (i.e., 100 years assumed for Class A and 500 years assumed for Class C), a factor of 10 increase applied to all Class C limits, and a factor of 22 increase in the Class A limit for Cs-137. To evaluate the site-specific waste class using the Category 3 approach of the NRC's concentration averaging guidance described in Section 3.5 and Appendix B of NUREG-1854, the DOE calculated the SOF based on the Class A limits using radionuclide concentrations decayed for 400 years. For radionuclides other than Cs-137, that method is equivalent to calculating a SOF with the Class C limits with undecayed radionuclide limits, if the factor of 10 increase in the Class C limits is removed. As described in NUREG-1854, it is appropriate to remove the factor of 10 increase in the Class C limits when using the Category 3 averaging approach because the factor of 10 accounts for features of the original scenario that are accounted for using site-specific parameters using the Category 3 averaging approach.

The CSSF also compared the calculated waste concentration for Cs-137 to the Class A value in Table 2 of 10 CFR 61.55. However, in the Hanford facility TER, NRC recommended that for WIR terminations, DOE should divide the Table 2 concentration for Cs-137 by 20 to account for adjustments made in the original rule for waste not being uniformly at the concentration limit in commercial LLW disposal. As a result, DOE underestimated the SOF for Cs-137 in its waste classification calculations for the CSSF. NRC performed independent calculations and confirmed that the SOF for both the CSSF bin sets and transport line waste profiles, including the adjustments to adapt the limits to the smaller volume of waste in the drilling scenario compared to a basement excavation scenario, thinner waste layer, and an additional 400 years of decay do not approach 1 for the Class A limits, which indicates that the undecayed concentrations do not exceed the site-specific Class C limits.

Waste classification is performed assuming the concentrations are known. However, the inventory of residual waste is an uncertain estimate based on limited sampling and assumptions in the ORIGEN2 model. The uncertainty in the radionuclide concentrations should be considered when classifying the waste. NRC does not expect waste generators to know the exact inventory of a bin set of waste before classifying it according to 10 CFR 61.55. Uncertainty is inherent in developing the inventory of the bin sets. DOE did not provide an uncertainty analysis specifically for the development of waste classification. NRC recommends that once DOE obtains more accurate waste inventory data, it may adjust waste classification calculations as necessary. It is important to note that the waste classification system was put in place to provide protection under 10 CFR 61.42. See Section 4.2.4 for additional information related to NRC's evaluation of the waste inventory. NRC recommends that DOE should perform a waste tank concentration comparison for inventories based on post-retrieval sampling once data becomes available.

The equations and inputs were clearly described and used the same methodology as the previously approved TER for the Draft WIR Evaluation for Closure of Hanford Site WMA C, with slight enhancements based on NRC recommendations in the TER. NRC agrees that DOE's conclusion that residual waste in the CSSF will be Class C or less.

4.2 Performance Assessment to Demonstrate Compliance with Performance Objectives

4.2.1 Summary of Performance Objectives

§ 61.40 General requirement.

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

§ 61.41 Protection of the general population from releases of radioactivity.

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

§ 61.42 Protection of individuals from inadvertent intrusion.

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

§ 61.43 Protection of individuals during operations.

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

§ 61.44 Stability of the disposal site after closure.

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

4.2.2 Performance Assessment Approach and Results

DOE presented results for the base case in the PA (DOE-ID, 2022). The simulations were updated following NRC staff's requests for additional information to change the thickness of the CSSF 1 vault to 0.6 m [2 ft] instead of 1.5 m [5 ft], and to change the Np-237 K_d from 500 L/kg to 100 L/kg. Three groups of radionuclides are simulated using the Mixing Cell Model computer code used for the near-field modeling, which only allows 10 radionuclides to be run at a time.

Group 1: H-3, Se-70, Tc-99, I-129 and Cs-135 (fission and activation products)

Group 2: Pu-240 (includes U-236, Th-232, Ra-228, and Th-228) and Np-237[4] (includes U-233, and Th-229) decay chain (actinide decay chain)

Group 3: Pu-239 (includes U-235, Pa-231, and Ac-227) and Pu-242 (includes U-238, U-234, Th-230, Ra-226, and Pb-210) decay chains (actinide decay chain)

DOE presented results for a "compliance time" (1,000 years) and extended out in time until the peak is reached. Peaks generally occurred well after the compliance period (>10,000 years). DOE stated that results beyond 1,000 years after closure are provided for information, but DOE warned that the results after 1,000 years should be viewed from an increasingly qualitative perspective.

Section 5.1.1.2 of the PA provides radionuclide fluxes from the base case analysis (DOE-ID, 2022). The flat portion of the curve presented in Figure 5-2 before 2,000 years for Tc-99 presents the minimum infiltration rate of 1×10^{-6} mm/yr [4×10^{-8} in/yr]. The cumulative flux is only 4.8×10^{5} Bq [1.3×10^{-5} Ci] of the total 1.5×10^{11} Bq [4 Ci], so most of the source term remains after more significant flow through the engineered system begins (i.e., the cementitious materials used to stabilize the waste are assumed to begin to fail at 2,000 years). Group 2 and 3 releases are much smaller and occur much later in time (10 thousand to 100 thousand years). DOE also noted that the early flat portions of U-233 and Th-239 are the minimum infiltration rates from in-growth from Np-237. U-236 is from both U-236 and the in-growth from Pu-240 (rolled up with the U-236 curve). For Group 3, U-235 fluxes include U-235 initial inventory plus in-growth from Pu-239. Additionally, U-238, U-234, and Th-230 come from themselves and in-growth from Pu-242.

Updated dose results are presented in Table 4-3 [see also Table 3 from DOE-ID (2025a)], which provides results from the original PA (DOE-ID, 2022) and updated results from re-running the base case with an updated concrete base thickness of 0.6 m [2 ft] and K_d of Np-237 of 100 L/kg. The peak dose occurs around 19,500 years due to the release of Tc-99 from CSSF 1. The peak dose is similar to the peak dose from the original PA because Tc-99 is long-lived, the source term is a near constant release, and a thinner concrete base only serves to slightly decrease travel time through the engineered system. The simulated tritium dose is actually lower due to less numerical dispersion in the thinner concrete base (the same number of elements were used to discretize the concrete base when the base case was re-run leading to less dispersion and consequently longer travel times for this relatively short-lived radionuclide).[5] However, the changes to the Base Case resulted in higher and earlier peak doses for Np-237 and primarily earlier peak dose for Pu isotopes albeit at levels significantly less than the Tc-99 doses.[6]

[4] Am-241 is stated to be run with Np-237. Because Am-241 has a 432-year half-life and decays to Np-237 which has a much longer half-life of 2.14×10^{6} years, almost all of the Am-241 will be converted to Np-237 within a few thousand years.

[5] Tritium is short-lived with respect to the travel time through the vadose zone.

[6] It is unclear why the long-lived radionuclide Pu-242 peak dose is significantly higher while there are insignificant changes in the doses reported for other plutonium isotopes in the revised Base Case.

Table 4-3: **Comparison of Base Case and Updated Base Case (CSSF 1 concrete base thickness and Np K_d were updated) Maximum Annual All-Pathways Effective Dose by Radionuclide and Time Period at the CSSF 1 Receptor Location. The highest dose for each radionuclide over time is bolded. Modified from Table 3 in DOE-ID, 2025a).**

Radionuclide	10,000-50,000 years mrem/yr	50,000-100,000 years mrem/yr	100,000-500,000 years mrem/yr	>500,000 yrs mrem/yr
Original Base Case				
Cs-135	1.46×10^{-11}	6.10×10^{-9}	1.13×10^{-4}	$\mathbf{4.01\times10^{-4}}$
I-129	$\mathbf{2.48\times10^{-4}}$	6.96×10^{-5}	7.18×10^{-5}	2.67×10^{-56}
Se-79	$\mathbf{2.19\times10^{-3}}$	1.07×10^{-3}	5.48×10^{-4}	1.25×10^{-32}
Tc-99	$\mathbf{1.91\times10^{-1}}$	4.07×10^{-2}	1.56×10^{-2}	0.00
Pu-239	6.08×10^{-9}	2.36×10^{-7}	4.67×10^{-6}	$\mathbf{1.05\times10^{-5}}$
Pu-240	4.96×10^{-9}	1.50×10^{-8}	8.05×10^{-6}	$\mathbf{1.86\times10^{-5}}$
Np-237	3.93×10^{-8}	2.51×10^{-5}	$\mathbf{2.03\times10^{-3}}$	1.52×10^{-3}
Pu-242	4.26×10^{-10}	1.57×10^{-7}	2.84×10^{-4}	$\mathbf{3.82\times10^{-4}}$
Total	$\mathbf{1.91\times10^{-1}}$	4.10×10^{-2}	1.61×10^{-2}	1.94×10^{-3}
Year	19,500 years	78,750 years	105,000 years	510,000 years
Updated Base Case				
Cs-135	1.52×10^{-11}	6.19×10^{-9}	1.13×10^{-4}	$\mathbf{4.01\times10^{-4}}$
I-129	$\mathbf{2.50\times10^{-4}}$	6.96×10^{-5}	7.18×10^{-5}	2.67×10^{-56}
Se-79	$\mathbf{2.21\times10^{-3}}$	9.30×10^{-4}	5.48×10^{-4}	1.25×10^{-32}
Tc-99	$\mathbf{1.92\times10^{-1}}$	4.07×10^{-2}	1.56×10^{-2}	0.00
Pu-239	3.50×10^{-8}	4.97×10^{-7}	$\mathbf{1.18\times10^{-5}}$	1.15×10^{-5}
Pu-240	3.09×10^{-8}	9.81×10^{-8}	$\mathbf{1.90\times10^{-5}}$	1.83×10^{-5}
Np-237	1.65×10^{-4}	5.11×10^{-3}	$\mathbf{5.57\times10^{-3}}$	1.32×10^{-4}
Pu-242	1.40×10^{-8}	4.03×10^{-6}	$\mathbf{9.02\times10^{-4}}$	8.40×10^{-4}
Total	$\mathbf{1.92\times10^{-1}}$	4.36×10^{-2}	2.15×10^{-2}	1.12×10^{-3}
Year	19,500 years	78,750 years	105,000 years	510,000 years
To convert mrem/yr to mSv/yr, divide by 100.				

Figure 4-1 shows the dose versus time at six receptor locations 100 m [328 ft] downgradient from each bin set. Although not clear from the tables and figures (because all sources contribute to dose at each receptor location), it appears that the CSSF 1 peak dose contributes most of the dose reported at receptor locations 1, 2 and 3 at about 19,500 years and that the peak CSSF 1 dose is significantly higher (around a factor of six) compared to the CSSF 2 peak dose, which occurs around 80,000 years. This is an important point because the results are dominated by CSSF 1 due to the assumptions regarding the shorter assumed time to failure of the CSSF 1 SS bin sets (see discussion in Section 4.2.5 of this TER). Less clear are the dose contributions of CSSF 3. The PA states that the CSSF 1 and CSSF 3 dose contributions combined are around 9 percent at around 80,000 years. However, since CSSF 2 and 3 have the same assumed times to failure of the SS bin sets and similar Tc-99 inventories (1.3×10^{11} and 9.3×10^{10} Bq [3.4 and 2.5 Ci] for Tc-99 in CSSF 2 and 3, respectively), it is unclear why there was no mention of the projected peak dose from CSSF 3 around the time of the CSSF 2 peak dose.[7]

An isopleth map of the doses from all CSSF bin sets at the time of peak dose is illustrated in Figure 5-14 of the CSSF PA (DOE-ID, 2022). This figure confirms that most of the dose at the time of peak dose (19,500 years) is from CSSF 1. Releases from the other CSSFs do not occur until later in time and although Tc-99 inventories are similar in all bins sets with a maximum inventory of 1.5×10^{11} Bq [4 Ci] in CSSF 1 and minimum inventory of 5.2×10^{10} Bq [1.4 Ci] in CSSF 4, because of the assumptions regarding SS bin set failure the doses from Tc-99 are significantly higher for CSSF 1.

The releases from CSSF 2-6 are much less significant compared to CSSF 1. As shown in Figure 8 of DOE-ID (2025a), groundwater annual all-pathways doses are dominated by Tc-99 (2 μSv [2×10^{-1} mrem]) followed by Se-79 (0.02 μSv [2×10^{-3} mrem]) between 10,000 and 50,000 years and by Np-237 and progeny (0.06 μSv [6×10^{-3} mrem]) after 100,000 years. The actinides (Pu-239, Pu-240, and Pu-242) and Cs-135 contribute little to the all-pathways dose until >500,000 years after the start of the simulation because these radionuclides are assumed to sorb strongly in the vadose zone and, consequently, have long unsaturated transit times. Figure 4-2 shows the key radionuclides contributing to dose at CSSF 1 for the updated base case.

One-Factor-at-a-Time Sensitivity Analysis Results

One-off sensitivity analyses (described as one-factor-at-a-time or OFAT analyses) are described in Table 4-4. The OFATs generally did not demonstrate the impact of failure of multiple barriers.[8] While the sensitivity runs appear to be pessimistic or bounding with respect to a single parameter, due to the redundancy of the barriers, the results may not show the full impact of potential underperformance of multiple barriers and the importance of individual barriers may be masked due to the presence of a redundant barrier. The one sensitivity run that considers a seismic “event” (i.e., potential “event” in a features, events, and processes evaluation) shows that the peak dose could be a factor of 36 times higher, suggesting the importance of the engineered system to performance. Most of the performance appears to come from the SS bins,

[7] The PA (DOE-ID, 2022) Section 5.1.3.1 indicates that most of the peak dose comes from CSSF 1 and 2 and implies that the dose from CSSF 3 is much less significant.

[8] OFAT 1 simulated earlier failure of the engineered system including the steel bins and the cementitious materials so really multiple factors were considered.

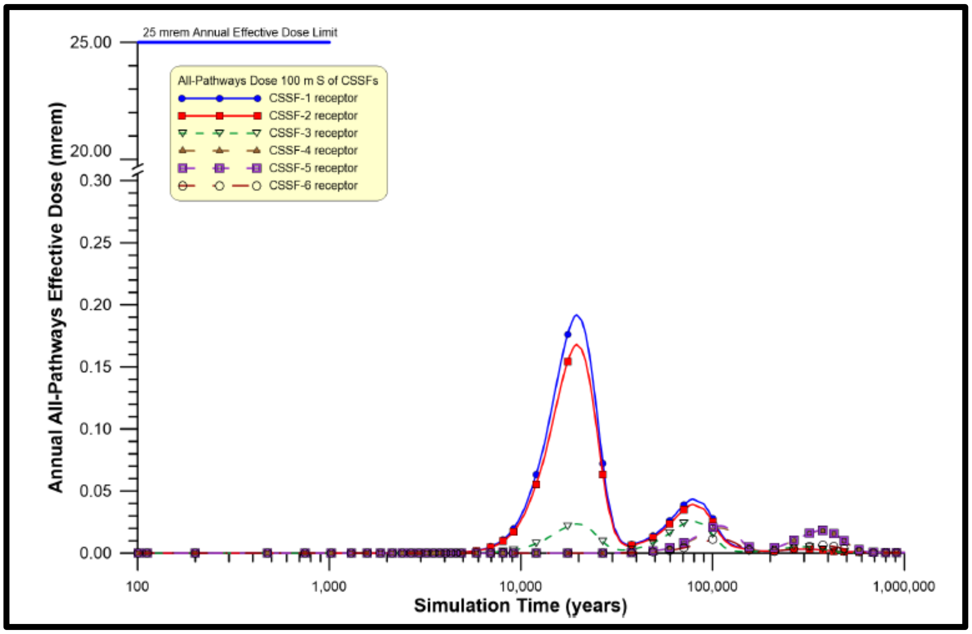


Figure 4-1: Updated Base Case All Pathways Effective Dose for Each CSSF 1-6 Receptor Location 100 m [328 ft] Downgradient of Each Source (Considers All Sources). Image Credit: Figure 7 DOE-ID (2025a). Replaces Figure 5-15 in the PA (DOE-ID, 2022).

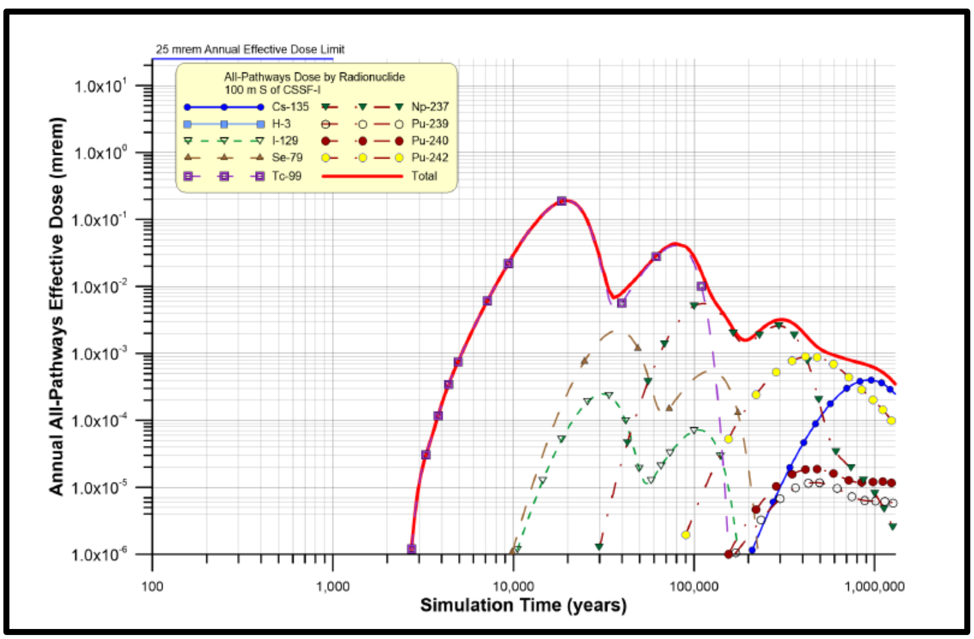


Figure 4-2: Rerun of the base case with 0.6 m [2 ft] thick concrete base for CSSF 1 and 100 L/kg K_d for Np in grout/concrete. Log-log plot of the groundwater annual all pathways effective dose by radionuclide for CSSF 1 receptor. Image Credit: Figure 10, DOE-ID (2025a). This figure replaces Figure 5-18 in the PA.

which outlast the cementitious materials, which are assumed to start failing around 2,000 years with sufficient degradation to allow the full 1 cm/yr [0.4 in/yr] infiltration rate by 3,300 years were it not for the SS bins limiting flow. Not only do the steel bins take long periods of time to degrade, the SS bins serve to mitigate the release of radionuclides from the grouted waste form by limiting infiltration through the system leading to very slow releases over time. The peak dose could be a factor of 31 times higher just due to the failure of the steel bins themselves. Although the infiltration rate in OFAT 4 was increased by a factor of 10, the peak dose only increased by a factor of 2 in sensitivity analysis. This result was unexpected, but the non-linear response was attributed by DOE to assumptions regarding cementitious material degradation and SS bin set corrosion, which dampened the effect of a higher infiltration rate. The highest dose resulted from a new analysis associated with OFAT 1, which included a higher infiltration rate and fracture flow through the cement base (represented by van Genuchten parameters for basalt and a factor of 10 lower K_d in the concrete base). The modified OFAT 1 analysis (OFAT 1c) resulted in a dose of 0.24 mSv/yr [24 mrem/yr], suggesting that the concrete base hydraulic and chemical performance could be important to dose when underperformance of the SS bin sets is assumed.[9]

While the OFATs were re-run with the thinner CSSF concrete base and lower Np-237 K_d, Figure 6-6 in DOE-ID (2022) was not reproduced with the updated results. Figure 4-3 below shows the PA results in DOE-ID (2022), which may not exactly match the results in Table 4-4 with the updated parameters. However, minor changes in the peak doses resulted from the updates to the parameter values due to the dominant radionuclide being long-lived and mobile Tc-99. In some cases (e.g., re-run of OFAT 2), the dose was even a little lower due to less overlap of the CSSF 1 and CSSF 2 doses with the decrease in CSSF 1 concrete base thickness.

As can be seen in Figure 4-3, even with the SS bins and grout assumed to be failed between 500-600 years, the release of Tc-99 (a nearly conservative species with limited to no sorption) takes hundreds of years to be released from the engineered system and to travel through the vadose zone (approximately 500 years). As discussed in more detail in Section 4.2.5, assuming the waste is not mixed in with the grout (i.e., the waste is assumed to be a discrete layer at the bottom of the bin sets), and assuming more discrete failure of the engineered system (i.e., higher steel bin failure rates) could lead to a significantly higher dose.

Groundwater Pathway Sensitivity and Uncertainty Analysis Results

The results of the uncertainty analysis show the importance of steel bin performance (and cementitious material performance) on the results. In particular, the steel bins are assumed to fail slowly over long periods of time (e.g., the bins take over 1 million years for complete failure due to corrosion). The percent failure of the bin sets (i.e., the percent of the bin area breached by corrosion) is an important parameter in the PA as the low infiltration rate of 1 cm/yr [0.4 in/yr] is reduced by the percentage of the bin set that is assumed to be failed to compute the water flux for radionuclide release (see Section 4.2.5 for more information).

[9] DOE discusses the impact of fracture flow and less sorption leading to differences in results between OFAT 1b and OFAT 1c.

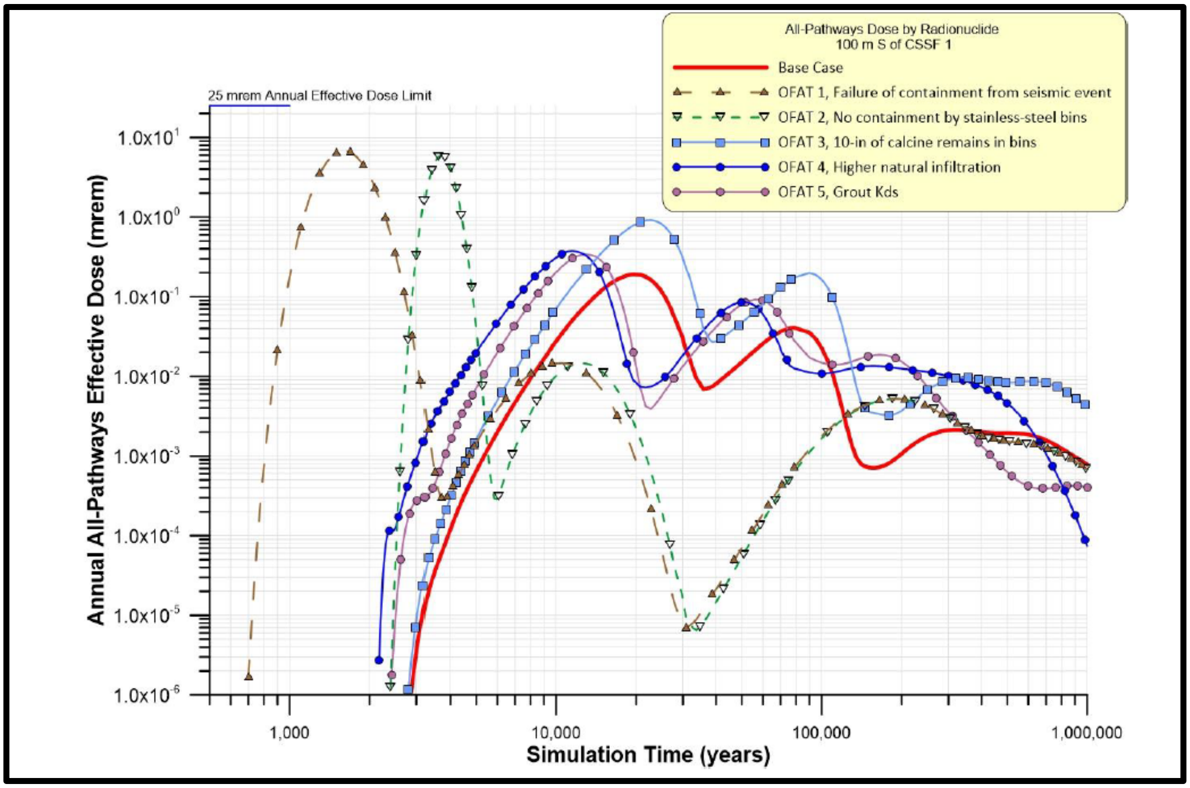


Figure 4-3: Groundwater Annual All-pathways Effective Dose for the Base Case and Original OFAT Cases 1-5. Image Credit: Figure 6-6 in DOE-ID (2022).

A parametric uncertainty analysis was also conducted using Monte Carlo sampling approaches combined with simple random sampling. Model sensitivity was evaluated by calculating the rank correlation between the output variable (e.g., groundwater concentrations and all-pathways doses) and the input parameter distributions.

Key assumptions include the following:

- Hydraulic properties of the vadose zone were held constant in the analysis
- Uncertainty in exposure scenarios was not evaluated
- Uncertainty in dose conversion factors was not evaluated
- The following radionuclides were considered: Tc-99, Se-79, and Np-237 (and progeny), because these radionuclides were thought by DOE to dominate the dose.

Uncertain parameters include the following:

- Infiltration rate after soil naturalization {maximum 2 cm [0.8 in] or twice base case values of 1 cm [0.4 in]}
- Geometric mean {CSSF 1 [5,226 to 140,000 years] and CSSF 2 [17,000 to 870,000 years]} and geometric standard deviation of the bin set lifetime (1.59 to 6.89)

Table 4-4: NRC Staff Description of Base Case and OFATs Used to Study Uncertainty in the CSSF Engineered and Natural System Performance.

Case	Description	RAI/CC	Result
Base Case	The Base Case assumes very low corrosion rates of SS bin sets (bins take over a million years to completely corrode and tens of thousands of years to reach a significant fraction of the long-term infiltration rate of 1 cm/yr [0.4 in/yr]) with releases modulated by the percent of the area of the steel liner failed. The waste is assumed to be mixed in the bottom 0.3 m [1 ft] of grout and the grout is assumed to be retained in the waste based on distribution coefficients for oxidized conditions (the impact of decreasing pH is not assessed in the Base Case). A relatively low, long-term infiltration rate of 1 cm [0.4 in] is assumed. No credit is given to sorption in basalt, so most of the natural system performance comes from the sedimentary interbeds and dilution in the Snake River Plain aquifer.	RAI-3 (rerun)	0.192 mrem/yr (19,500 years)
1	Seismicity and accidents and unplanned events—A seismic event at 500 years resulting in failure of the containment structure consisting of both grouted CSSF bin sets and SS bins. Complete loss of containment over 100 years (hydrologic failure of the grout and complete corrosion of the bins 600 years after closure). Leaching of waste at 500 years at 1 cm/yr [0.4 in/yr]	Not applicable	6.82 mrem/yr (1000-2000 years)
1a	Rerun of 1 (re-run with 0.6 m [2 ft] concrete base for CSSF 1 {instead of 1.5 m [5 ft] for CSSF 1} and intended Np-237 K_d of 100 L/kg instead of 500 L/kg).	RAI-3	6.9 mrem/yr (1,300 years)
1b	Same as 1/1a with enhanced infiltration (i.e., 18 cm/yr [7.1 in/yr])	CC-5	9.94 mrem/yr (550 years)

Table 4-4: Description of Base Case and OFATs Used to Study Uncertainty in the CSSF Engineered and Natural System Performance. (cont'd)

Case	Description	RAI/CC	Result
1c	Same as 1b with enhanced infiltration and fracture flow through the concrete base (i.e., the same van Genuchten parameters are used for fractured basalt and reduction of the sorption coefficients by a factor of 10)	CC-5	23.6 mrem/yr (525 year)
2	Extreme case with no credit for the SS (grout still has to degrade over time)	Not applicable	5.96 mrem/yr (3,675 years)
2a	Rerun of 2 (re-run with 0.6 m [2 ft] concrete base for CSSF 1 {instead of 1.5 m [5 ft] for CSSF 1} and intended Np-237 K_d of 100 L/kg instead of 500 L/kg).	RAI-3	5.79 mrem/yr (NR)
3	Inventory, radionuclide and other materials—25 cm [10 in] of calcine remain instead of 5 cm [2 in], but the inventory is mixed in 0.6 m [2 ft] instead of 0.3 m [1 ft] of grout.	Not applicable	0.919 mrem/yr (~20K years)
3a	Rerun of 3 (re-run with 0.6 m [2 ft] concrete base for CSSF 1 {instead of 1.5 m [5 ft] for CSSF 1} and intended Np-237 K_d of 100 L/kg instead of 500 L/kg).	RAI-3	0.920 mrem/yr (NR)
4	Climate change, regional and local {increase in precipitation and 10 times higher infiltration rate of 10 cm/yr [4 in/yr]}	Not applicable	0.377 mrem/yr (11,500 years)
4a	Rerun of 4 (re-run with 0.6 m [2 ft] concrete base for CSSF 1 {instead of 1.5 m [5 ft] for CSSF 1} and intended Np-237 K_d of 100 L/kg instead of 500 L/kg).	RAI-3	0.374 mrem/yr (NR)
5	Chemical, geochemical processes and conditions (K_d for sediment substituted for grout if it was lower)	Not applicable	0.338 mrem/yr (~13K years)
5a	Rerun of 5 (re-run with 0.6 m [2 ft] concrete base for CSSF 1 {instead of 1.5 m [5 ft] for CSSF 1} and intended Np-237 K_d of 100 L/kg instead of 500 L/kg).	RAI-3	0.339 mrem/yr (~13K years)
5b	Same as 5a with reduced K_ds in cementitious materials (stage 4/oxidized) and in natural system (lower of grout or screening level)	RAI-4	0.34 mrem/yr (12,500 years)

Table 4-4: Description of Base Case and OFATs Used to Study Uncertainty in the CSSF Engineered and Natural System Performance. (cont'd)

Case	Description	RAI/CC	Result
6	Impact of Tc-99 flux from thinner waste form {5.1 cm [2 in] of waste mixed in 10.2 cm [4 in] of grout {instead of 30.5 cm [12 in] of grout} for Tc-99 only}.	Not applicable	Tc-99 flux was 1.35 times higher.
7	Upper-bound corrosion rates of SS suspended in calcine {28 to 50 × higher than the base case corrosion rates—0.3 versus 0.01 μm/yr [1×10^{-5} versus 4×10^{-7} in/yr] for CSSF 1}	RAI-3 (related to RAI-2)	2.92 mrem/yr (3,525 yr)
7a	Same as 7 with pulse of water through the waste zone {water comes through a hole in the top after 2,000 years and is stated to rapidly transmit through shrinkage gaps and accumulate the bottom of the bin; the accumulated water drains over a 10 year period at 18.5 cm/yr [7.3 in/yr] for CSSF 1 and 28 cm/yr [11 in/yr] for CSSF 2, then goes back to 1 cm/yr [0.4 in/yr] after 10 years}[10]	RAI-3	4.06 mrem/yr (2,700 years)
8	Screening K_ds for the natural system	RAI-4	0.193 mrem/yr (19,500 years)
Multiple-Factors-at-a-Time (MFAT)	Pessimistic performance of all barriers. SS bins in calcine, grout fails at 500 years, higher infiltration rate of 10 cm/yr [4 in/yr], concrete base fractures (van Genuchten parameters for fractured basalt and minimum/oxidized sorption coefficients), minimum sorption for grout (waste layer) from either Hanford Tank PA (oxidized conditions) or Savanah River Site Tank Farm PA (Stage III oxidized conditions), pessimistic K_ds in the natural system (lower value of the screening level or the grout K_d).	RAI-6	11.1 mrem/yr (775 years)
Note: NR = not reported. To convert mrem/yr to mSv/yr, divide by 100.			

[10] See Table 5 in DOE-ID (2025a) for estimated corrosion rates and times for SS in a calcine storage environment.

- Longevity of the grout (maximum of 2,600 years or twice the base case value of 1,300 yrs)

- Time of failure of the grout (700 to 2,000 years with 2,000 years being the base case value)

- K_d values in grout (Tc, Se, Np, U and Th)

- K_d values in sedimentary interbeds (Tc, Se, Np, U and Th)

- Radionuclide inventory (factor of 2 different for Tc, Se, Np)

- Aquifer flow and transport parameters {dispersivity factor of 2 different from base case value of 3.31 m [10.86 ft]; Darcy velocity had a minimum of 15, a mode of 21.9, and maximum of 25 m/yr [49, 72, and 82 ft/yr, respectively]

Parameter distributions are listed in Table 6-3 of the PA (DOE-ID, 2022). The results of the uncertainty analysis are provided in Figure 4-4. DOE did not update the probabilistic analysis in response to RAIs (addressed RAIs with OFATs). The timing of the peak dose in the PA is between 3,000 and 4,000 years (DOE-ID, 2022). To address NRC's RAI comment about the potential risk dilution, DOE indicated that the mean of the peak dose is less than a factor of two higher than the peak of the mean (DOE-ID, 2025a). Additionally, the deterministic dose {2 μSv/yr [0.2 mrem/yr]} occurs much later and is lower than the peak of the mean and 95th percentile of the probabilistic dose of 6.4 μSv/yr [0.64 mrem/yr] and 22.9 μSv/yr [2.29 mrem/yr], respectively. Based on Table 6-5 in the PA (DOE-ID, 2022) of the radionuclides studied (Se-79, Tc-99, Np-237, U-233, and Th-229), a significant portion of the dose is associated with Tc-99 followed by Np-237 and U-233, with the 97.5 percentile doses of Np-237 and U-233 still contributing orders of magnitude less than Tc-99.

The results of the analysis suggest bin lifetime is negatively correlated to dose, Tc-99 K_d in interbed and grout are negatively correlated to dose, and there is a positive correlation of Tc-99 inventory to dose. There is a strong positive correlation between geometric standard deviation (GSD) of bin failure times to dose. DOE explains on page 6-26 of the PA that the positive correlation for the annual all-pathways dose at 4,050 years shown in Figure 6-9 of the PA (DOE-ID, 2022) is due to the greater likelihood of a release occurring at earlier times with a greater GSD, although peak releases will be smaller.

DOE provided sufficient information to allow NRC staff to assess the impact of engineered and natural system performance on the demonstration of compliance with the performance objectives in 10 CFR Part 20. A listing of the base case and OFAT analyses and results is provided in Table 4-4. The results show that the performance objective of 0.25 mSv/yr in 10 CFR 61.41 for protection of the general population from releases of radioactivity are met for all cases, even those cases that DOE characterizes as not credible.

Due to the redundancy of barriers, a more robust analysis of the impact of assumed underperformance of engineered barriers and/or more pessimistic assumptions regarding natural system performance were assessed by NRC staff to inform its decision, recommendations, and potential future monitoring of the CSSF. See Sections 4.5, 4.6, and Appendix A for more information.

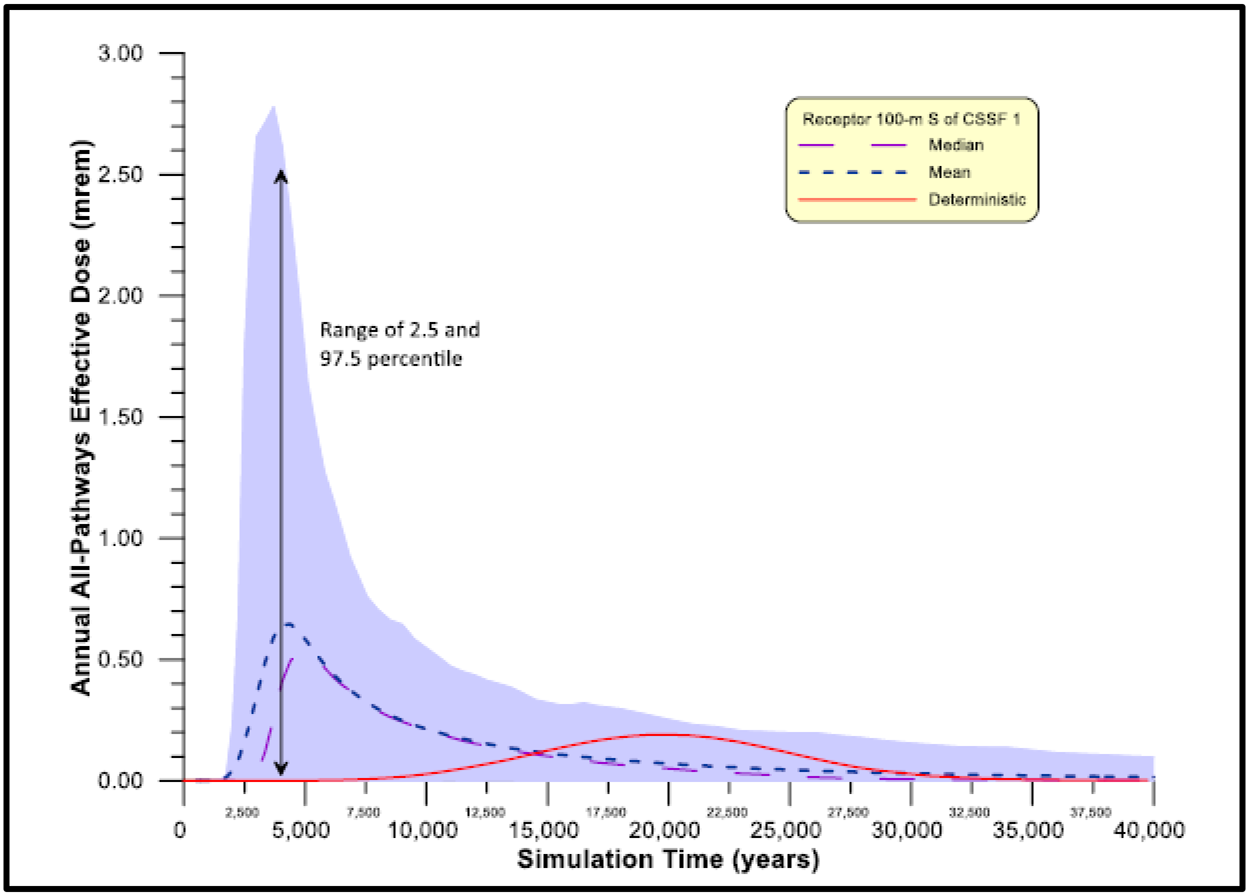


Figure 4-4: DOE's Probabilistic All Pathways Effective Dose Results for CSSF 1 Receptor. Image Credit: Figure 6-7 (DOE-ID, 2022).

4.2.3 Infiltration and Erosion

Infiltration and erosion are discussed in Sections 3.2.2.7 of the PA (DOE-ID, 2022). The closure strategy has not been finalized, although Figure 3-4 (DOE-ID, 2022) illustrates the assumed end state of the CSSF for the purposes of the PA. Figure 3-4 shows the current configurations of CSSF 1-3 located underneath a berm and CSSF 1-6 protruding above grade. The PA does not include an engineered cover; therefore, NRC staff were not able to evaluate potential effects of an engineered cover on infiltration. For the purposes of the PA, infiltration and erosion are assumed to be representative of historical to long-term trends.

DOE-ID (2022) uses infiltration estimates in two contexts: (i) calculation of releases from the engineered system and (ii) transport within the unsaturated zone of the natural system. NRC staff evaluate the use of the infiltration estimates for calculating releases in Section 4.2.5 but note that infiltration into the soil immediately adjacent to CSSFs extending above ground may be locally enhanced by runoff from the top of the CSSF or enhanced snowpack accumulation adjacent to the shaded north side of the CSSF.[11]

The following evaluation is specific to net infiltration as it relates to transport of releases through the unsaturated zone (i.e., not for processes inside the CSSF engineered barriers). In this context, net infiltration affects the travel time for radionuclides to transit the unsaturated zone.

DOE-ID (2022) represents the infiltration rate in the CSSF vicinity using two infiltration estimates. The initial infiltration rate is assumed to be 18 cm/yr [7.1 in/yr] based on the analyses of precipitation infiltration at INTEC, an area with gravel topsoil, summarized in the Operable Unit (OU) 3-14 Remedial Investigation/Baseline Risk Assessment (RI/BRA) (DOE-ID, 2006). DOE-ID (2022) assumes that soil naturalization begins after 50 years and takes 50 years to complete. Soil naturalization includes the establishment of vegetation and soil development. Naturalization is assumed to gradually reduce net infiltration to 1 cm/yr [0.4 in/yr] consistent with natural infiltration of 0.36 to 1.1 cm/yr [0.14 to 0.43 in/yr] estimated adjacent to the north boundary of the Radioactive Waste Management Complex (RWMC) under undisturbed conditions (Cecil et al., 1992).

The infiltration calculations used for the PA analyses are based on site estimates of net infiltration (i) at several locations in the vicinity of INTEC (DOE-ID, 2006) and (ii) adjacent to the RWMC (Cecil et al., 1992). Cecil et al. (1992) describe the soil adjacent to the RWMC as silty loam (a fine-textured soil), while DOE-ID (2006) describes the soil at INTEC as coarse and gravelly. Construction drawings and discussion (DOE-ID, 2025b) provided in response to NRC questions during the RAI response public meeting on June 26, 2025, suggest the excavated material was likely used for backfill around each CSSF and the berm around CSSFs 1 through 3 was compacted. Construction boreholes for CSSFs 2, 5, 6, and 7 consistently describe the soil zone as gravelly to sandy, many with a 0.5 to 5 m [1.6 to 16 ft] layer of clay or clay-rich material above the basalt. The RWMC area is relatively undisturbed. The INTEC area is disturbed ground with no vegetation, ample hard cover (buildings and pavement) promoting runoff and focused flow with losses from unlined ditches, and has experienced various leaks during operations, all potentially contributing to historical observed perched water under the site.

Without a final closure strategy, NRC staff consider these estimates as hypothetical infiltration rates under the assumption that no cap system will be in place. Under this scenario, the provided information raised potential technical issues including the following: (i) soil texture differences between the RWMC and INTEC sites may foster larger net infiltration at the INTEC site than at the RWMC site even after vegetation establishment, and (ii) the assumed soil naturalization rate is unrealistically rapid. The following discussion explains the technical issues and addresses the NRC staff rationale for concluding that DOE provided sufficient information to address the potential issues.

NRC staff recognize that net infiltration is difficult to accurately estimate in semiarid environments. Net infiltration represents a small difference between gains (precipitation) and losses (runoff, evapotranspiration), which all exhibit substantial uncertainty. Infiltration pulses are typically the result of sporadic events (or a series of events) that are large enough to allow moisture to rapidly penetrate to depth below the evapotranspiration zone (e.g., snowmelt events); these may only occur for a few years in a decade, leaving few opportunities to measure

[11] DOE-ID (2022, page 3-19) indicates that the infiltration rate into the top of protruding CSSFs may be lower due to lack of accumulation of snow on top of the CSSFs.

fluxes. For a given magnitude of infiltrating water, a soil with a coarser texture will tend to have a smaller change in moisture content than a soil with a finer texture, thus the penetration depth tends to be larger with coarser soil textures and more of the infiltration escapes below the evapotranspiration zone.

Reynolds and Fraley (1989) studied rooting depths at the INL site and determined that the maximum rooting depth is 225 cm [89 in] for the dominant plant species, big sagebrush. Cecil et al. (1992) observed a chlorine-36 bulge in the top 1 to 1.5 m [3.3 to 4.9 ft] of the soil column, a tritium response to 4 m [13 ft], and fluctuating water content in neutron probe access holes past 5 m [16 ft] in depth, suggesting that wetting pulses penetrated at least most of the way through the root zone at the RWMC site but only a small fraction of water passed entirely through the rooting zone.

Any difference in soil texture between the two sites would be especially important during cool seasons, when soil water can redistribute to depth while vegetation is dormant and potential evapotranspiration demand is small. For example, snow tends to begin accumulating at INL starting in November, then several months of accumulated snow may melt during a relatively short period in February or March as a distinct snowmelt event, perhaps coupled with a rainfall event (Clawson et al., 2018). In three years since the early 1950s, the snowmelt event occurred on a frozen surface that caused flooding episodes (Clawson et al., 2018). In other years, the snowmelt event may provide a wetting pulse (akin to a ponded infiltration test) that may move well below the rooting depth before the native vegetation is active.

NRC staff thinks that soil texture differences between the RWMC and INTEC sites may foster larger net infiltration at the INTEC site than at the RWMC site even after vegetation establishment. DOE-ID (2006) provides several examples at the INTEC site where water was observed penetrating rapidly to depth when supplied with water at the surface. For example, a ponded infiltration test with shallow water at the northwest corner of INTEC showed a wetting front penetrating >3 m [>10 ft] in 2 hours and 5.8 m [19.0 ft] in 20 hours. Depositing fine-textured soil over the coarse soil will hold water close to the surface, but it is unclear how much fine-textured soil is necessary to substantially dampen wetting pulses.

DOE provided evidence that the coarser texture may not lead to large net infiltration, even without vegetation present. Soil moisture data from monitoring wells at the Remote Handling LLW facility, provided in response to NRC clarifying comment 7 (DOE-ID, 2025a), showed a small transient moisture response over a month in 2 of 9 wells during the 2019 snowmelt period and little to no response in the subsequent two years. No estimates of infiltration are available for the observations. The site is described as backfilled with gravel. Unfortunately, the observation period did not experience a wet winter. NRC extracted cumulative winter precipitation (defined as 9/30 through 3/30) for each year from the precipitation data from nearby Arco, ID, which has a record starting in 1913 (more than three times longer than the record from the local Mesonet station located at INTEC). The 2019, 2020, and 2021 winter periods evaluated in the response had 1.12, 0.37, and 0.28 times the median winter precipitation. The nine winters from 2013 through 2021 included the driest 2 through 6 winters of the 99 on record at Arco (2020 and 2021 were ranked 4 and 3 driest, respectively). NRC interpreted the data during this recent time period as suggesting that perching above the basalt would not occur in most years but not ruling out the possibility of perching occurring in wet years.

The PA represents flow rates entering the vadose zone as annual rates that slowly change over time, even though infiltration is seasonal (e.g., events during cool and wet periods). In the response to RAI PA6, DOE-ID (2025a) calculated the impact of cyclically concentrating annual average infiltration into three consecutive months, mimicking a seasonal infiltration from seasonal snow melt and the subsequent relatively wet months. The calculated magnitude of cyclic variations around the average flow rate decayed considerably with depth in the basalt. A hypothetical source at the top of the basalt exposed to the flow showed cyclic mass flux variations near the top of the basalt, but the mass flux arriving at the base of the basalt was essentially unaffected by the variations. NRC accepts that mass flux rates will strongly dampen with depth when the travel time is much longer than the interval between events, but NRC expects actual infiltration to be distributed as less frequent but much larger events (e.g., almost all infiltration flow that reaches the basalt layer occurs over a few weeks after a snowmelt event for an especially wet winter), with dynamics consistent with the rapid time scales of event data from the Remote Handling LLW monitoring wells provided in response to NRC clarifying comment 7 (DOE-ID, 2025a). Large pulses may result in a system that bypasses upper interbeds during the infrequent intervals with large flow. This would effectively reduce the overall travel times and retardation through the vadose zone and likely not eliminate the contributions from lower in the vadose zone.

Transport of radionuclides through the vadose zone only occurs after releases have occurred, so infiltration rates become significant to transport after steel bins in the CSSF have developed penetrations and have begun to release radionuclides. DOE-ID (2022) expects that significant hydrologic failure of the bin sets will require a minimum of tens to hundreds of thousands of years, much longer than the hypothesized 100 years to closure and 50 years to re-establishment of natural soil and vegetation proposed in the PA. Section 4.2.5 evaluates PA modeling assumptions that determine infiltration rates through the engineered system.

4.2.4 Radionuclide Inventory

Inventory development is discussed in Section 2.3 through 2.5 of the PA. Section 3.1 of this TER focuses on inventory development for the purpose of demonstrating compliance with Criterion 2 but is also relevant to development of the inventory for the 10 CFR 61.41 evaluation.

Because inventory development is already discussed in Section 3.1, this section of the TER focuses on use of the inventory to perform radionuclide screening analyses focused on the groundwater pathway. Screening analyses for the groundwater, air, and intruder pathways is discussed in Sections 2.6 through 2.8 of the PA (DOE-ID, 2022).

DOE performed a screening analysis to reduce the initial list of 148 radionuclides to a more manageable number based on risk. DOE first screened the list using a half-life cut-off of 5 years. The basis for the 5-year half-life cut-off is that only 1 millionth of the inventory will remain after the 100-year institutional control period, thereby significantly reducing the risk of the short-lived radionuclides. After the Phase 1 screening, 75 of the 148 radionuclides in the original inventory remained. However, two of the isotopes were not radioactive (i.e., were stable nuclides) and two were noble gases and not important to the groundwater pathway. After removing these four radionuclides, 71 radionuclides were retained for Phase II screening. Even though some of the radionuclides were short-lived, if a parent isotope were present, then the ingrowth of the progeny was considered in the GWSCREEN Phase II screening. Figure 2-49 in the PA provides the conceptual model for GWSCREEN Phase II screening (DOE-ID, 2022).

A simplified geometry and Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) Track 2 risk assessment parameters, which were stated to provide pessimistic estimates of groundwater concentrations and ingestion doses at the INL site, were used in the second GWSCREEN screening. The source geometry was assumed to be equivalent to a 33×33 m [108×108 ft] square area. The 5.1 cm [2 in] of remaining calcine waste is assumed to be mixed in 0.3 m [1 ft] of grout.

A number of assumptions were made in the GWSCREEN modeling that DOE characterized as being conservative. These assumptions include (i) the selection of the more conservative K_d between cementitious or natural system materials, (ii) use of a 10 cm/yr [4 in/yr] infiltration rate through the source zone, (iii) lack of consideration of transport through basalt layers {i.e., only 14 m [46 ft] sedimentary interbed was considered}, and (iv) other simplifications {see PA Section 2.6.2 for additional details [DOE-ID, 2022]}.

After Phase II screening, 12 of 71 radionuclides remained. Additional radionuclides were also included for other reasons (e.g., U-235, U-236, and U-238 were retained because the chemical concentrations are important for regulatory compliance). Parent to Am-241 and Np-237, the Pu-241 screening dose was only 24 μSv/yr [2.4 mrem/yr], but Pu-241 may contribute to its daughter doses and was therefore included. Th-230 was also retained because it has a screening dose of 22 μSv/yr [2.2 mrem/yr] or more than half the screening dose limit and is one of the progeny of the radionuclides of concern, U-234. A final list of 17 radionuclides was retained in the PA inventory (see Table 4-5).

The total volume of calcine assumed to be accumulated in the solids transport lines is about 845 times smaller than the total volume of 25.4 cm [10 in] of residual calcine in the bins assumed for screening and is about 170 times smaller than the volume from 5.1 cm [2 in] of residual calcine in the bins, as assumed for the groundwater pathway base case. There are 613 m [2,011 ft] of piping; however, only 24 m [79 ft] of piping is assumed to have potentially accumulated radioactive calcine (EDF-11119; DOE-ID, 2018a). During the RAI response meeting held on July 26, 2025, DOE agreed to provide the transfer line inventory, which was only used in the inadvertent intruder calculations discussed in Section 4.3 of this TER. See DOE-ID (2025b) for the transfer line inventory.

NRC requested additional information from DOE regarding its screening analysis, which was based on a 40 μSv [4 mrem] dose for individual radionuclides and only considered the drinking water pathway. NRC also questioned the conservatism of the analysis (e.g., credit for engineered or natural system performance). DOE ran additional screening analyses with all pathways and lower dose limits and determined that additional radionuclides could have been screened in (Nb-94, Sn-126, and U-236). DOE provided justification for the conservatism of its screening analysis in lieu of including additional detailed analysis for Nb-94 and Sn-126 (U-236 was already included with its decay chain). While NRC staff noted other more conservative analyses which did not credit any decay during transport, thereby screening in short-lived radionuclides such as Sr-90 with the highest activity and highest potential hazard in the CSSF, the approach used by DOE for the CSSF is reasonable. NRC staff also performed its own independent modeling to determine if additional radionuclides could be screened in and the impact of uncertainty on the results of the PA analyses. This information is used to make recommendations to DOE as well as to inform any potential future monitoring of the CSSF, as applicable.

Table 4-5: Select Screened in Radionuclides (Based on Half-Life and GWSCREEN Calculations). Modified from Table 2-26 in DOE-ID (2022). See Additional Detail in Table 2-24 (DOE-ID, 2022) on K_ds.

Radionuclide[a]	Half-Life (year)	K_d Grout (L/kg)[b]	K_d VZ (L/kg)
Am-241(Np-237)	4.32×10^{2}	2 (Np-237) [b]	2 (Np-237)
Cs-135	2.30×10^{6}	1	50
H-3	1.23×10^{1}	0	0
I-129	1.57×10^{7}	0	0
Np-237	2.14×10^{6}	2	2
Pu-238 (U-234)	8.77×10^{1}	1.6 (U-234)	1.6 (U-234)
Pu-239	2.41×10^{4}	22	22
Pu-240	6.56×10^{3}	22	22
Pu-241 (Np-237)[c]	1.43×10^{1}	2 (Np-237)	2 (Np-237)
Pu-242	3.73×10^{5}	22	22
Se-79	1.10×10^{6}	0.1	4
Tc-99	2.11×10^{5}	0	0
U-238[d]	4.47×10^{9}	1.6	1.6

a. Progeny are noted in parentheses.
b. Relatively short-lived Am-241 is assumed to be retained in the waste zone where it transforms immediately to its relatively long lived and mobile daughter Np-237. The transport rate of Np-237 dictates the release and transport through the vadose zone.
c. Although the maximum screening dose is <40 µSv/yr [<4 mrem/yr], Pu-241 decays quickly to Am-241 and then Np-237, it could be a contributor to the Np-237 dose and was retained in the analysis.
d. Although the maximum screening dose is <40 µSv/yr [<4 mrem/yr], U-235, U-236, and U-238 were retained because besides the dose comparison, the Calcined Solids Storage Facility performance uranium chemical concentrations to the contaminant level for regulatory compliance.

NRC recommends that DOE consider sampling of the residual waste after waste retrieval to provide a better estimate of the remaining CSSF inventory following closure. Sampling would help validate the assumptions[12] that went into development of the inventory [see RAI PA-6 in NRC (2024)]. Composite sampling could be used to ensure an adequate number of locations to estimate the mean inventory in the bin sets while reducing analytical costs associated with analyzing individual increments in each composite sample.

4.2.5 Near-Field Modeling

This section evaluates DOE's assumptions regarding the hydraulic performance of the cementitious materials, and the SS bin sets in limiting flow through the waste and out of the engineered system. The SS bin sets are particularly risk-significant due to the large uncertainty associated with their long-term hydraulic performance, particularly given the complexity associated with modeling the flow of infiltrating water through the waste zone and engineered

[12] Assumptions include (i) use of a limited number of liquid waste samples prior to calcination or ORIGEN modeling to estimate the inventory, (ii) use of scaling factors for hard-to-detect (HTDs), (iii) residual volumes based on expected waste retrieval operations, and (iv) assumption that residual waste is a homogenous mix of the various waste streams that went into the bin sets, among other potential uncertain assumptions.

system as the SS bin sets corrode over time.[13] The near-field section also evaluates releases from the engineered system into the natural environment.

Engineered Barrier Modeling

The cementitious material degradation depends heavily on the analysis in the INTEC TFF which considered (i) sulfate and magnesium degradation, (ii) carbonation, and (iii) calcium hydroxide leaching and simple comparisons between the engineered systems. However, the final INTEC TFF PA adopted a conservative assumption (i.e., the PA assumed engineered barriers failed at 500 years despite much longer failure times estimated by detailed analyses) and, therefore, the staff found the INTEC TFF cementitious material degradation modeling acceptable for the purposes of the PA.

The assumptions in the CSSF modeling include the following:

- The water flux through the vault wall and grout is initially set based on the saturated hydraulic conductivity of intact grout as shown in Figure 3-11 of the INTEC TFF PA (DOE-ID, 2003) at 10^{-13} m/s [3×10^{-8} ft/day] for Zone b.

- At 2,000 years, the grout between the vault wall and SS bin begins to degrade. As the grout degrades, hydraulic conductivity increases logarithmically to 10^{-6} m/s [0.3 ft/day] which occurs at approximately 5,000 years after the start of the simulation [Figure 3-11 in the INTEC TFF PA (DOE-ID, 2003)].

- The PA model assumed that the infiltration rate starts increasing from 2,000 years until equating the natural net infiltration rate at the soil surface of 1 cm/yr [0.4 in/yr] after 1,300 years, or at 3,300 years from the start of the simulation.

The PA indicates that the grout outside the SS bins (between the vault wall and bins) and concrete base will degrade long before the SS bins fully fail, and that grout inside the bins are pessimistically assumed to evolve and change like the grout between the vault wall and the SS bins.

OFATs considered early and more rapid cementitious material failure, fracture flow through the concrete base, and waste mixed in a thinner waste zone (see Table 4-4); as well as Monte-Carlo analyses to evaluate the impact of uncertainty in the time of the onset of grout degradation, and the time of full grout degradation.

Corrosion Modeling

The corrosion model is discussed in Section 4.1.2.2 of the PA (DOE-ID, 2022) and considered three periods

- Period 1: filling of bins with calcine to removal of calcine from the bins (72-year duration)
- Period 2: grouting of bins to full degradation of grout (full grout degradation approximately after 15,000 years)
- Period 3: post-grout failure

[13] Flow of infiltrating water through the waste zone is important to projected dose.

Period 1 was modeled by computing the corrosion damage accumulated over 72 years, considering constant corrosion rates of SS equal to 6.78×10^{-4} mm/yr [2.67×10^{-5} in/yr] for 405 SS (bin material of CSSF-1) and 2.2×10^{-4} mm/yr [8.7×10^{-6} in/yr] for 304 and 304L SS (bin materials of CSSF-2, 3, 4, 5, and 6). The cumulative corrosion damage, assumed to occur on both sides of the bin (inner and outer side) was subtracted from the minimum bin thickness to define the bin thickness at the time of closure (after calcine removal). The corrosion rates for 405, 304, and 304L SS {6.78×10^{-4} and 2.2×10^{-4} mm/yr [2.67×10^{-5} and 8.7×10^{-6} in/yr]} were based on average values from dry-air oxidation tests using SS coupons suspended above the calcined solids inside CSSF-2 to CSSF-6, two-month dry-air oxidation experiments with SS coupons suspended above simulated calcine (non-radioactive) at 150 °C [302 °F], and two-month humid-air oxidation experiments with 304L SS coupons in contact with simulated calcined solids.

DOE-ID analysts combined periods 2 and 3 into a single effective period with a unique corrosion rate. DOE considered that alkaline porewaters controlled by grout during Period 2 would passivate the SS, with very low corrosion rates during that period. The rate of corrosion during Period 3 would be expected to be higher, and DOE considered reasonable to use the same corrosion rates for Periods 2 and 3, based on corrosion rates applicable to Period 3. The corrosion rates assumed in the model for this combined period were based on SS coupons exposed to soil near the CSSF up to 12 years (Adler-Flitton et al., 2011). DOE considered that the soil environment would bound the grout porewater environment (because steel becomes passivated in alkaline grout porewater). During the 12-year soil exposure, Type 304L SS coupons exhibited negligible corrosion damage; in many instances the damage was below the resolution of the weight-loss technique used to establish corrosion rates. A total of 14 non-zero corrosion rates for 304L SS were extracted from the multiple weight loss soil exposure measurements. The modelers postulated corrosion rates follow a log-normal distribution and used the 14 data points to compute a geometric mean {3.64 nm/yr [1.43×10^{-7} in/yr]} and a geometric standard deviation 2.38 (dimensionless) [Table 4-5 in DOE-ID (2022)]. This log-normal distribution spans 2 orders of magnitude between the 0.004 and the 0.996 quantiles of the distribution.

The non-zero measured corrosion rates were only a few nanometers per year, well below passive corrosion rates on the order of microns per year commonly reported for SS when measured with electrochemical techniques in aqueous solutions. Using the distribution of corrosion rates, the modelers computed a distribution of failure times assuming the bins corrode simultaneously from the inside and outside of the bin at the same corrosion rate. The corrosion rates for CSSF 1 were increased by a factor 3, because CSSF 1 bins are made of Type 405 SS, and experiments exposing 405 SS coupons to calcine were up to three times higher than corrosion rates for Type 304 and 304L SS.

The modelers interpreted the cumulative distribution function (CDF)[14] of the failure times as a curve representing gradual damage versus time of the SS bin surface due to corrosion. The coupons exposed to soil would represent a fraction of the bin surface; thus, the failure time computed with an empirical corrosion rate measured with coupons would represent the time it takes a portion of the SS bin to be penetrated by a corrosion front.

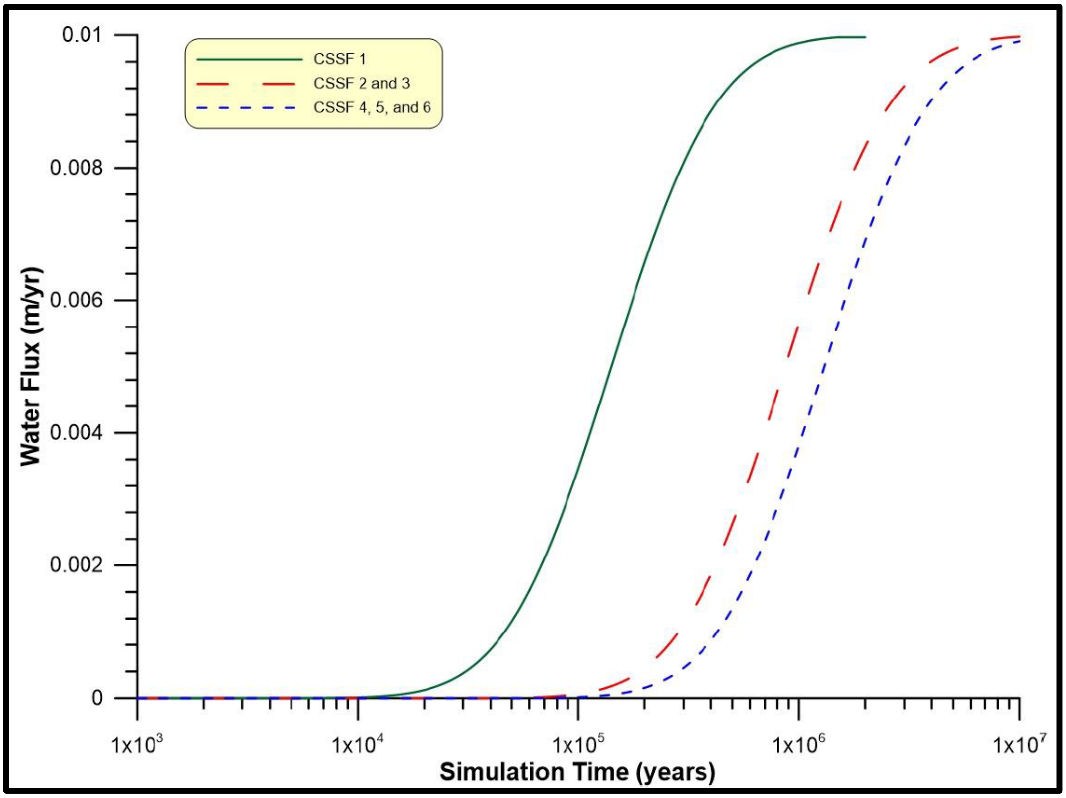
[14] See Equation 4-43 of DOE-ID (2022) for calculation of the CDF based on the geometric mean and geometric standard deviation of the soil corrosion data.

Water flux through the SS bins mobilizing radioactivity was computed based on a simple model, including two factors. One factor of the water flux was a log-linear ramp function of time, associated with grout degradation [Figure 3-7 of DOE-ID (2022)]. The second factor was the bin failure time CDF, related to gradual degradation of the steel bin surface due to chemical corrosion. The failure time CDF was used to multiply the grout water fluxes to establish the fluxes and flow rates that could mobilize radionuclides in the calcine zone, after the gradual corrosion of the SS bins (i.e., the infiltration rate through the waste zone was modulated by the percent area of the steel liner breached as a function of time based on the failure time CDF), with complete bin surface failure oftentimes taking millions of years and median (i.e., 50 percent of the bin surface) failure times of 100,000s to millions of years (see Figure 4-5). The hydraulic conductivity of the grout used to stabilize the waste could also limit flow; however, in the Base Case the hydraulic conductivity of the grout is high enough to allow the 1 cm/yr [0.4 in/yr] background infiltration rate at 3,300 years. On the other hand, at 3,300 years, a negligible area of the steel bin surface is assumed to have been corroded and therefore the SS bin greatly limits flow through the system to negligible levels for tens of thousands of years.

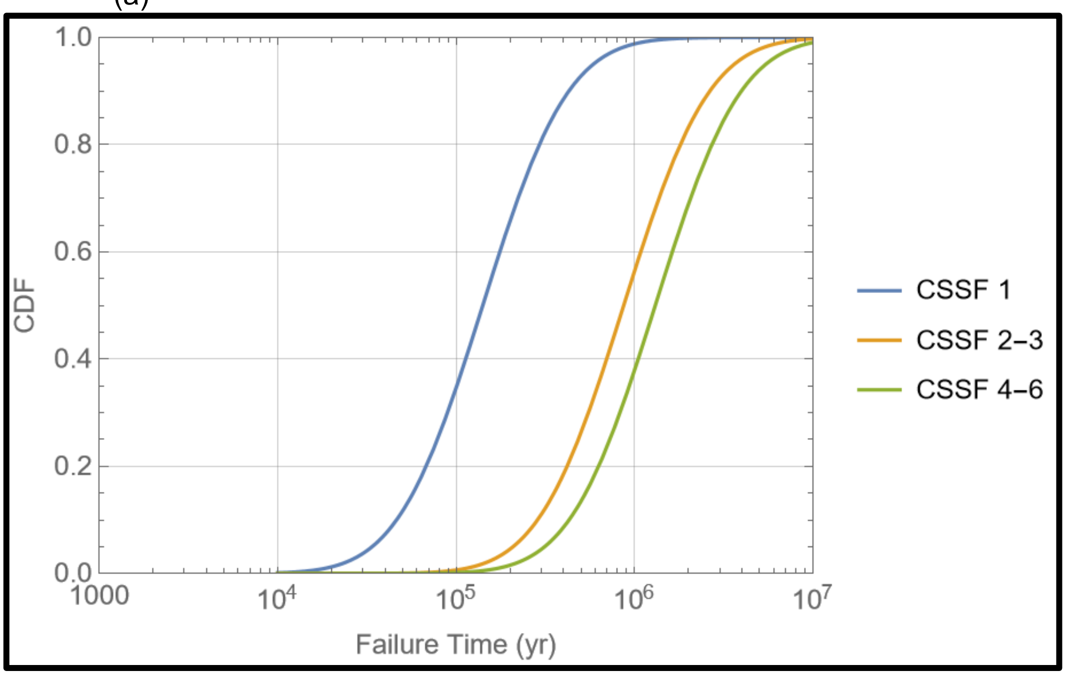
Figure 4-5 compares the water fluxes and the failure time CDF. Figure 4-5(a) reproduces Figure 5-1 of DOE-ID (2022); the plot represents the water flux through grout versus time, computed considering two factors previously described. The Figure 4-5(b) is the failure time CDF computed assuming corrosion rates follow log-normal distributions. The curve shapes demonstrate that the bin failure time CDF factor is dominant to determine the water flux versus time.

NRC staff requested additional information to support DOE’s assumed SS corrosion rates and modeling approach (NRC, 2024). The corrosion rates of the Base Case assumed to apply to long time frames after removal of the calcine and after grouting were based on data collected from corrosion coupons exposed to natural and unprotected soil over a 12-year period (Adler-Flitton et al., 2011). Because of the arid climate, the natural soil is expected to be relatively dry and not promote corrosion. In fact, many SS test 304L coupons did not exhibit measurable corrosion over the 12-year period (Adler-Flitton et al., 2011). Non-zero corrosion rates were only detected in a few coupons. These non-zero corrosion rates are summarized in 14 data points in Table 4-5 of DOE/ID-12008 (DOE-ID, 2022) that were used to develop a distribution of corrosion rates for the PA model. DOE’s estimated corrosion rates were on the order of a few nanometers per year, that are consistent with passive corrosion rates. NRC requested additional information to support the approach used to calculate corrosion rates following closure (i.e., SS in contact with grout), including an evaluation of alternative mechanisms that could lead to higher corrosion rates (e.g., crevice corrosion, pitting corrosion, galvanic corrosion). Additionally, NRC staff had technical issues with the conceptual model that assumed the fraction of the bin area that is breached directly controls the fraction of flow through the engineered system mobilizing the radioactive waste (e.g., 100 percent of the bin area would have to be breached to allow 100 percent of the background 1 cm/yr [0.4 in/yr] infiltration rate to flow through the waste zone). This conceptual model is discussed further in the “Waste Release Modeling” and “Uncertainty Analysis” sections below.

In its RAI, NRC staff stated that the soil exposure testing was not designed to represent a set of conditions that would unequivocally “bound” corrosion in pristine or degraded grout, including corrosion in crevices that might accumulate chemicals promoting localized corrosion. Corrosion rates in the literature for SS/grout systems, for example measured with electrochemical methods using aqueous solutions, are commonly higher than the nanometer per year corrosion rates measured in 12-year soil exposure using weight loss experiments.



(a)



(b)

Figure 4-5: (a) Water Fluxes Through the Grouted Waste Layer as a Function of Time; Reproduced from Figure 5-1 in DOE-ID (2022). (b) CDF of the Corrosion Failure Times, Assuming Corrosion Rates Follow a log-normal Distribution.

In response to NRC's RAI, DOE noted that information from literature does not represent the service environment for the CSSF bin sets. Studies in the literature commonly measure corrosion rates in aqueous solutions with high concentrations of chloride. By contrast, only minute concentrations of chloride are expected in the infiltrated water and in groundwater. DOE stated that grouts will be prepared with low chloride content to ensure low chloride concentrations in grout porewater. Nonetheless, to address the question of uncertainty in exposure conditions and uncertainty in corrosion rates, DOE performed additional analyses assuming higher corrosion rates based on measurements using coupons suspended over calcine and in contact with simulated calcine. Additionally, DOE performed additional OFAT or MFAT simulations to examine consequences of degraded barrier capability performance.

DOE qualitatively examined enhanced corrosion mechanisms such as pitting and crevice corrosion, and stress-corrosion cracking. DOE concluded those enhanced corrosion processes are unlikely because the required environmental conditions (including high chloride concentrations) are not expected to arise, and also because DOE expects water contacting the SS would be conditioned by the grout thereby passivating the SS. Test coupons suspended above the calcined solids extracted from the CSSF bins did not show any evidence of localized corrosion. Sensitized SS and welds may exhibit higher corrosion rates. DOE noted that the fraction of the bin surface covered by welds is small (DOE-ID, 2025a). According to DOE arguments, it is expected that welds would also exhibit passive behavior, due to the expected alkalinity of water and porewater conditioned by grout, and the low concentration of chloride expected in the water. Coupons suspended above calcine for several years and extracted from bins exhibited low corrosion rates. These coupons were exposed to bin temperatures which could have sensitized the SS material of the coupons (DOE-ID, 2025a); however, the extracted coupons exhibited negligible corrosion. DOE did not model welds differently than the base SS material of the bins. Nonetheless, DOE executed OFAT simulations to examine uncertainty in corrosion rates of bin materials.

NRC staff notes that DOE assertions stating that soil exposure is more aggressive than exposure to grout porewater can be misleading, as well as assertions that corrosion rates based on soil exposure are conservative. NRC staff agrees that soil exposure may be more likely to exhibit signs of enhanced corrosion rates such as pitting corrosion. However, the arid climate limits the time infiltrating water would contact the SS coupons, and it is not evident that a dry environment would "bound" wet environments like grout porewaters. Nonetheless, the soil exposure coupons and the suspended coupons over calcined solids indicate very low corrosion rates, and that localized corrosion is unlikely in the CSSF. The low corrosion rates measured after 12 years of soil exposure (Adler-Flitton et al., 2011) may reflect the combined action of humid air corrosion, aqueous corrosion during brief periods, and atmospheric corrosion. The soil exposure corrosion rates implicitly incorporate the effect of weather at the location of the CSSFs, which may be a reasonable consideration in the PA model.

The Center for Nuclear Waste Regulatory Analyses (CNWRA®) performed field experiments using SRS tank grout formulations to study the potential for preferential flow pathways to form in grouted tank systems. These experiments showed that gaps may develop after grout shrinkage at interfaces with steel tanks leading to potential bypassing flow pathways through the grouted tanks (Dinwiddie et al., 2011).[15] CNWRA also performed experiments to study the conditioning

[15] The CNWRA was tasked by NRC to provide information on the physical and chemical degradation of cementitious waste forms used for the isolation and containment of radioactive wastes. Additionally, CNWRA evaluated the potential for radionuclide bypassing of engineered barriers through preferential or

of infiltrating groundwater using SRS reducing grout formulations (Walter and Dinwiddie, 2021). The E_h in the experiments decreased to values significantly higher than assumed in DOE SRS's PAs, while pH increased to strongly alkaline conditions during the experiment (and then slowly started to decrease). In DOE SRS's models, the E_h increases, and the pH decreases as a function of the number of pore volumes that pass through the system. If flow were to occur rapidly through preferential pathways, then the level of groundwater conditioning by the grout may be limited (e.g., the pH of water contacting steel may not be controlled by the grout). DOE-ID has not specified a grout formulation to be used to stabilize waste in the bins, and in particular if a reducing grout will be designed. Therefore, findings from the CNWRA experiments may or may not apply to CSSF.

DOE has indicated that SS is passivated in alkaline solutions. Also, DOE indicated that infiltrated water is of low chloride concentration and would not promote SS corrosion. It is unlikely enhanced corrosion modes (e.g., crevice or pitting corrosion) would arise in alkaline solutions of low chloride concentration; however, there is a level of uncertainty with respect to the evolution of the water chemistry in grout shrinkage gaps or annuli and their effect on the corrosion of SS bin walls contacted by the water.

Aggressive environmental conditions may develop if the calcined solids become wet. However, there is no evidence that the bins have been compromised during their service, or that the calcined solids have become wet, or that the calcined solids have decomposed and released corrosive chemicals, as revealed by the coupons suspended over the calcined solids (Hoffman, 1975). Removed coupons from the bins have shown minor corrosion, likely due to dry air oxidation, without any indication of localized corrosion. Residual calcined solids might become wet during the time the bulk of the calcined solids will be removed, and the bins are grouted. Eventually the chemistry of water in contact with the SS bin materials is expected to initially be controlled by the grout (e.g., due to the presence of bleed water from grouting) until the grout cures. There is uncertainty in corrosion mechanisms during the time water chemistry transitions from control by the calcined solids to control by the grout porewater. Yoder (2001) exposed 304L SS coupons to simulated calcined solids (non-radioactive) for two months at 32 °C [90 °F] in a chamber controlling the relative humidity to 60 percent, and measured minimal corrosion of the SS. These experiments suggest that calcines in humid air may not

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fast flow pathways. Review of DOE's WDs and PAs indicate that potential fast flow pathways through, or around, grout barriers may dominate waste release from grout-filled tanks and vaults (Walter et al., 2009). CNWRA's mesoscale and intermediate-scale grout monolith experiments provided data to assess the significance of various types of cracks and thermal contraction shrinkage gaps in grout, including along-wall annular gaps, horizontal lift separations, and grout flow lobe shrinkage gaps. Gaps may develop between the cementitious grout and the containment walls, as well as between individual grout flow lobes or lifts placed at different times. Shrinkage gaps were observed to develop early during the hydration and aging process at interfaces between both INL and SRS waste-stabilizing grouts and the containers in which these grouts were placed (Walter et al., 2009, 2010; Dinwiddie et al., 2011). Through-going cracks and along-wall shrinkage gaps or annuli can provide direct, fast flow paths from the top to the base of a waste form. In such cases, infiltrating meteoric water may not be exposed to the grout long enough to be chemically conditioned by it. DOE prefers to model slow plug flow of water through the grout matrix in grouted waste forms, during which residence time allows the grout to evolve the water chemistry towards alkaline conditions. However, fast preferential flow of meteoric water through cracks and gaps in the grout may be more realistic and result in minimal chemical conditioning. Grout-component water-conditioning tests, grout component mixture water-conditioning tests, and grout water-conditioning tests performed by CNWRA (Walter and Dinwiddie, 2019, 2020, 2021) illustrated how the pH of aqueous solutions in contact with individual grout components, grout component mixtures, and tank grout evolve over the course of immersion experiments lasting up to 170 days.

release chemicals corrosive to 304L SS, assuming the simulated calcine is representative of the actual calcine waste. Nonetheless, there is a level of uncertainty in corrosion rates during transient times calcined solids may become wet.

To explore uncertainty in corrosion rates, DOE considered case OFAT 7 in an RAI response (DOE-ID, 2025a), which assumed higher constant corrosion rates of the SS bins than the base case applied immediately after closure. These higher corrosion rates were based on test coupons suspended above calcined solids for 17 years, and two-month test exposing coupons to simulated (non-radioactive) calcined solids. The assumed median corrosion rates were 0.304 μm/yr [1.20×10^{-5} in/yr] (CSSF-1), 0.183 μm/yr [7.20×10^{-6} in/yr] (CSSF-2, 3, and 4), and 0.207 μm/yr [8.15×10^{-6} in/yr] (CSSF-5 and 6), and corrosion rates spanned a factor 69.5 (CSSF-1), 4.13 (CSSF-2, 3, and 4), and 2.7 (CSSF-5 and 6) between the median and the 0.99 quantile of the assumed log-normal distribution. DOE doubled the corrosion rates to account for simultaneous outside-in and inside-out corrosion acting on the SS bin. In the RAI response, DOE stated that DOE did not consider the OFAT 7 case to be credible but provided the information to address questions on the robustness of the closure system. DOE concluded that the maximum effective dose associated with case OFAT 7 was well below an annual effective dose limit of 0.25 mSv/yr [25 mrem].

In the Base Case, DOE assumed the water flux contacting the waste at the bottom of the SS bins is linearly proportional to the extent of corrosion degradation of the bin surface. NRC considers this conceptual model to be unsupported. Once the grout degrades, flow contacting intact portions of the bin top must locally build pressure above the bin to drive laterally diverting flow over the bin, potentially allowing radial flow to converge into openings in the bin top. In case of focused flow paths, a fractional opening of the bin surface may be sufficient to fully capture the infiltrated water. The NRC posed RAIs aimed at DOE examining focused flow paths, and the possibility that only a fraction of the degraded SS bin surface may be sufficient to allow the full focused flow to contact and mobilize the waste (NRC, 2024). In response to RAI PA-3, DOE provided additional OFAT cases or PA simulations (DOE-ID, 2025a):

OFAT 1a: case with premature failure of bins, grout, and concrete

OFAT 1b: same as OFAT 1a with enhanced infiltration

OFAT 1c: same as OFAT 1c with enhanced infiltration and fracture flow in the concrete base

OFAT 2a: no credit for SS bins

OFAT 7: higher SS bin corrosion rates, based on SS coupon exposure to calcine and simulated calcine

OFAT 7a: 10-year pulse flow starting at 2,000 years after closure, followed by infiltration modulated by SS bin corrosion damage after 2010 years, with corrosion rates as in case OFAT 7.

The results of these analyses are discussed further in the “Waste Release Modeling” section below.

DOE did not examine the possibility of galvanic corrosion arising between dissimilar metals. DOE stated that bins were made of single-type SS alloy (DOE-ID, 2022). Dissimilar metals may come into contact at transfer line inlets at the top of the bins. DOE generically described these transfer lines to be made of carbon steel and SS (DOE-ID, 2022). In the case of carbon steel pipes, carbon steel may preferentially corrode and protect the SS material of the bins. The SS of transfer pipes is likely different than the SS alloy of the bins, and a level of galvanic corrosion may arise. However, pipes inside vaults will also be grouted and expected to exhibit passive behavior due to the alkalinity that DOE expects and low chloride concentration of the surrounding water chemistry. Nonetheless, there is uncertainty in corrosion rates, especially of dissimilar metals in electric contact. It is possible that joints of dissimilar metals may corrode sooner, although the relative surface of those joints at the top of the bins is a tiny fraction of the total bin surface and far from the location of the residual calcine waste.

To further examine how uncertainties in the corrosion rates affected flow through the waste, DOE developed a special scenario in case OFAT 7a, considering early water penetration at the top of the bin until the inside grout pore volume was filled. DOE assumed the water would be fully flushed during a 10-year period after full degradation of the grout (starting 2,000 years after closure), due to a later opening in the bottom of the bin. The flux required to fill the internal bin pore volume in 2000 years would be greater than the natural background infiltration of 1 cm/yr [0.4 in/yr]. After the flushing process, the case OFAT 7a assumed that fluxes would return to being modulated by the fraction of the corroded bin surface. The case OFAT 7a also assumed higher corrosion rates of the bin materials, based on coupons suspended over calcined solids. NRC staff thinks that this case adequately evaluates the potential effect of higher corrosion rates at joints at the top of bins (which might be associated with galvanic corrosion of dissimilar metals), and higher corrosion rates at the bottom of the bins (which might be associated with different grout water chemistries due to the presence of residual calcine or crevice corrosion). See “Waste Release Modeling Section” for more details on the impact of underperformance of the engineered system on waste release. NRC staff consider that DOE reasonably evaluated scenarios of enhanced chemical degradation of the SS bins in response to RAIs and as part of cases OFAT 7 and OFAT 7a. For example, 0.99 quantile of the corrosion rate distribution was as high as 21 μm/yr [8.3×10^{-4} in/yr] for CSSF-1, as high as 12.7 μm/yr [5.0×10^{-4} in/yr] for CSSF-2, 3, and 4, and as high as 14.4 μm/yr [5.7×10^{-4} in/yr] for CSSF-5 and 6. Those upper-bound corrosion rates represent enhanced corrosion modes, which address uncertainties of corrosion mechanisms and corrosion rates of SS bins. However, the analysis does not address conceptual issues with the use of the CDF to moderate releases through the bin sets based on the assumed fraction of the bin sets that is corroded over time. See “Waste Release Modeling Section” for additional details.

By contrast, the 0.99 quantile corrosion rates of the Base Case were equal to 0.76 μm/yr [3.0×10^{-5} in/yr] for CSSF-1 and 0.25 μm/yr [9.8×10^{-6} in/yr] for CSSF-2 to 6, which are corrosion rates of magnitude consistent with passive corrosion rates measured with electrochemical methods in aqueous solutions reported in the literature. DOE provided information in the OFAT cases to examine consequences of higher corrosion rates of SS bins, which allowed NRC staff to evaluate barrier robustness. DOE examined the relevance of the SS bins to prevent and limit radionuclide releases in other OFAT cases discussed in the Waste Release Modeling section below.

An important aspect of the engineered barrier system in the PA analysis is DOE’s assumed long lifetime of the SS bins, which is dependent on (i) high pH and low chloride concentration environments, (ii) low chloride concentration environments after grout is degraded, and (iii) passivity of SS (which requires the formation of passive oxide films on the SS surface). As part of the closure strategy, NRC staff recommends designing a monitoring system to track the chemistry of infiltrated waters and environments in contact with the grout, and to monitor degradation of SS materials. Placing probes in contact with the SS bins may not be desirable;

probes and grout openings may have uncertain long-term effects. A safer alternative may be designing coupons of SS to mirror conditions of actual SS bins, surrounded by grout and simulated calcined solids. Electrochemical probes could be attached to the coupons to track parameters such as corrosion electric potential and corrosion currents to verify passivity assumptions. The chemical composition of moisture and any potential deposits around coupons may be tracked to measure pH and chloride concentrations. The grout-steel coupons may be placed in soil in orientations designed to maximize contact with infiltrating water. The grout may be designed with cracks and gaps to explore effects of grout defects on infiltrating water and potential development of deposits in cracks and grout gaps. These efforts would help strengthen the assumptions regarding DOE's assumed long lifetime of SS bins.

Waste Release Modeling

Section 4.1 of the PA (DOE-ID, 2022) discusses the groundwater model, including the near-field conceptualization. As stated in the previous section, barriers to release include the SS bins, grout, and concrete vault. Over time, the concrete vault and grout are assumed to degrade, allowing water to flow through the structure. The SS bins are also a barrier to flow and are assumed to corrode slowly over time (long after the concrete and grout are expected to degrade based on the INTEC TFF PA modeling).[16] Although the entire bin and vault will be filled with grout, only the top 1.5 m [5 ft] of grout is modeled. The waste is assumed to be mixed in the bottom 0.3 m [1 ft] of grout with a 1.2 m [4 ft] layer of clean grout cover above. The PA model discretized the grouted bin into five mixing cells representing the 1.5 m [5 ft] of grout. The lowest 0.3 m [1 ft] of the cell is where the waste layer resides (cell 5 from the top). Leaching of radionuclides from the contaminated grout is controlled by the assumed K_d of grout that is assumed to be oxidized.

The 1.5-m [5-ft]-thick concrete base consists of four 0.3-m [1-ft]-thick grid cells [see Table 4-7 of DOE-ID (2022)]. The base of each bin set was assumed to reside at the alluvium/basalt interface. Leachate from the concrete base flows directly to the unsaturated zone. Figures 3-3 and 3-4 from the PA provide information about the assumed bin set configuration including height of the concrete vaults above ground surface and depth of waste. Figure 3-5 from the PA provides a conceptualization of the release from the engineered system (DOE-ID, 2022).

The bottom of the concrete base provides the interface between the source model and the vadose zone model. The vadose zone model is assumed to gain water at the current infiltration rate (which is larger than the loss from the source term model until the grout and bin steel are sufficiently degraded) and is assumed to gain radionuclide mass at the same rate that the source term model loses mass (the mass flux rate is conserved). Figure 3-10 from the PA shows that the water flux throughout the vadose zone model reaches equilibrium with a long-term infiltration rate of 1 cm/yr [0.4 in/yr] after 150 years, while Figure 37 shows that the long-term flux rate through the concrete base is less than 1 mm/yr [0.04 in/yr] before 3,000 years and reaches 1 cm/yr [0.4 in/yr] after 3,300 years.

The Mixing Cell Model (MCM) code was used to model leaching of radioactivity from the source zone to the vadose zone, which was also modeled using MCM. MCM is a 1-D numerical model for transient water flow and contaminant transport in partially saturated media. MCM

[16] Figure 3-8 in the PA (DOE-ID, 2022) shows that at 5,000 years, the flow rate is reduced by the steel bin alone (the grout is completely failed at 3,300 years); however, it takes >>10,000 years for the full, assumed long-term infiltration rate of 1 cm [0.4 in] to be realized. At the time of peak dose (19,500 years), the infiltration rate is a small fraction of the peak long-term infiltration rate of 1 cm/yr [0.4 in/yr] (see Figure 4-5).

incorporates material-specific moisture characteristic curves that describe the relationship among moisture content, hydraulic conductivity, and pressure. Radionuclide fluxes from the source zone model were read into the vadose zone simulation as an upper-boundary condition.

No solubility control is assumed in the waste zone. Only K_ds reflective of oxidized conditions are used to simulate leaching from the waste zone, with the assumed inventory located in the bottom 0.3 m [1 ft] of grout. Tables 4-6 and 4-7 list the assumed values for cementitious materials and for the natural system used in the PA (DOE-ID, 2022). The natural system values are discussed in Section 4.2.6 of this report.

As can be inferred from the results of the OFAT scenarios 1 and 2, more rapid and complete failure of the SS bins may be the most significant challenge to performance of the CSSF. Additionally, the difference in travel times through a 0.6 m [2 ft] concrete base for CSSF 1 and the assumed 1.5 m [5 ft] concrete base assumed in the PA may be insignificant when releases are spread out in time with steel degradation assumed to occur over long time periods[17] in the Base Case, which leads to a relatively continuous source term with respect to the travel time to the point of exposure. However, the assumed performance of the grout and underlying concrete base may be more risk-significant for rapid events occurring over decades to centuries (i.e., a pulse release).

Therefore, NRC staff requested additional information on consideration of alternative and reasonable bounding scenarios to address engineered barrier performance uncertainties. As discussed above, the near-field model assumes flow through the waste zone is limited by the fraction of SS bin area that is assumed to be corroded, and potential reduction in flow through the calcine layer while the monolithic grout degrades.[18] A nominally low, long-term infiltration rate of 1 cm/yr [0.4 in/yr] is also assumed after vegetation is assumed to be reestablished (after 100 years). All of these assumptions serve to limit the projected flow through the system and limit modeled waste release.

Given uncertainty in long-term performance and absent condition-specific empirical data on SS degradation, NRC requested that DOE evaluate what it considered reasonable scenarios with degraded performance of the bins soon after the concrete structures and grout degrade, especially near the base of the bins, to account for uncertainties in corrosion degradation processes. For example, an alternative scenario with a more rapid pulse release of waste from the engineered system was requested to address uncertainty in engineered barrier performance.

NRC also requested additional information regarding the geometrical specifications of the bin sets and potential for void space in the bin sets following grouting. DOE/ID-12008 (DOE-ID, 2022) does not describe the geometry of the bin bottoms and it was unclear (i) whether the bin bottoms are flat or rounded, (ii) the area of contact between the steel and base, and (iii) the degree that grout can penetrate between the steel and concrete base. In response, DOE provided additional geometrical details on bin set construction. The bins typically have rounded bottoms; cylindrical skirts are welded to the bins to provide stands so that the bins can be rested on end. The skirts have cutouts to allow air flow under the bins.

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[17] CSSF 1 bins take more than 1,000,000 years to completely fail and other CSSFs take up to 10,000,000 years to completely fail. Significant flow through the system does not occur for tens of thousands of years.

[18] In the base case, the grout is assumed to be hydraulically degraded prior to significant steel bin failure.

Table 4-6: K_ds for Grout and Concrete (L/kg). Modified from Table 4-11 of DOE-ID (2022).

Element	Best-Estimate	Minimum	Maximum
Cs	1	0.1	10
H	0	0	0
I	3	0.3	30
Np	500[19]	100	25,000
Pa	100	71	140
Pb	500	360	710
Pu	100	71	140
Ra	50	5	500
Se	6	0.1	400
Tc	1	0.7	1.4
Th	30,000	1,000	1,000,000
U	2,000	1,400	2,800

Table 4-7: K_ds for Sedimentary Interbed and Basalt (L/kg). Modified from Table 4-12 of DOE-ID (2022).

Element	Interbed	Basalt
Cs	500	0
H	0	0
I	3	0
Np	18	0
Pa	500	0
Pb	270	0
Pu	1,140	0
Ra	500	0
Se	4	0
Tc	0.1	0
Th	500	0
U	10	0

There appears to be a potential for void space to remain underneath the bin sets (and under the top of the bins within the bin space) after grouting due to poor access and the potential for air trapping. For similar reasons, the thin annular air gaps between concentric bins in CSSF 1 may be difficult to completely grout and therefore may allow a permeable flow pathway. Void space may be partially mitigated through use of a flowable grout formulation, although more flowable grout may lead to poorer grout quality. Flowable grout typically has a large water content, which (i) may lead to excessive bleed water, (ii) may be vulnerable to shrinkage during curing, and (iii) may lead to cured grout with relatively large porosity and permeability.

As described in the “corrosion” section above, DOE conducted additional OFAT analyses to address technical issues identified in NRC staff’s RAIs (NRC, 2024); OFAT 7 was created to address comments regarding engineered barrier performance considering what DOE describes

[19] Note that DOE intended to use a value of 100 L/kg for Np-237 and provided updated results in response to NRC staff’s RAIs (DOE-ID, 2025a).

as upper bound corrosion rates. OFAT 7a, consider the higher corrosion rate in OFAT 7 with a pulse of water flowing through the calcine waste layer over 10 years after degradation of the grout and concrete base. The results of OFAT 7a are higher than the Base Case of 2 μS/yr [0.2 mrem/yr] at around 40 μS/yr [4 mrem/yr] but still less than the performance objective of 250 μSv/yr [25 mrem/yr]. DOE also noted that the OFAT 1 results were higher than the OFAT 7 results and OFAT 1 was therefore considered bounding.[20]

While NRC staff agree that the OFAT analyses provided useful insights on the importance of engineered barriers to the results, as well as potential dose due to underperformance of the engineered and natural system barriers, a more systematic barrier analysis could have provided additional insight. Additionally, an alternative conceptual model that assumed more discrete failure of the SS bins (compared to prolonged releases due to the use of the SS bin set CDF to moderate the infiltration rate) would have been informative to address conceptual model uncertainty on the results.

With regard to sorption coefficients, NRC had comments regarding the approach used to select values for the screening and detailed analyses, development of parameter distributions, and lack of consideration of changes in sorption coefficients over time as the grout degraded. DOE responded to NRC's requests for additional information and performed additional analyses to address uncertainty in sorption coefficients. Of note, the Np-237 sorption coefficient of 500 L/kg used for grout was stated to be in error. All analyses were re-run with the corrected of 100 L/kg for Np-237 as described in Table 4-4.

With regard to NRC staff technical issues concerning the lack of consideration of degradation of the cementitious materials and lowering of pH, DOE indicated that the transition to oxidized region III (ORIII) conditions (where pH may eventually be lowered to a value of around 8) is not risk-significant, and that the K_ds it selected reflected oxidizing conditions, which was more risk-significant. DOE also indicated that the pH of INL groundwaters was relatively high at a value of 8 and that it was unclear that ORIII conditions would ever be achieved. NRC staff notes, however, that the INL INTEC groundwater pH of 8 is at the lower end of the pH range for ORIII conditions, so while it may take longer for the pH to decrease, grout components will continue to be leached, and the pH should eventually drop to a pH of about 8 reflective of equilibration of pore water with calcite.

DOE ran additional OFAT analyses with more pessimistic values for the cementitious material sorption coefficients. The original OFAT 5 in the PA used lower or similar cementitious material K_ds compared to the base case for moderately oxidizing conditions (DOE-ID, 2025a, Table 12). In response to the NRC comments, DOE performed OFAT 5b, which used even lower cementitious material K_ds for Pu, Ra, and I reflective of Stage 4 conditions (DOE-ID, 2025a, page 54). OFAT 5b also included some reductions in natural system K_ds. The results indicated that using more pessimistic cementitious material K_d values did not result in doses exceeding

[20] OFAT 1 results, which simulate faster times to hydraulic failure of the SS bins and grout, are also higher than OFAT 2 results, which assumes that the SS bins are completely failed. While OFAT 2 would seem to address the issue with the assumed performance of the SS bin sets, OFAT 2 does not address other NRC staff technical issues (e.g., NRC staff expect that the waste may not be mixed in the grout, the grout may not retain key radionuclides, and flow may occur through bypassing preferential pathways around the grout and through the waste zone).

performance objectives, although the dose increased for several isotopes studied.[21] The impact of more conservative K_ds for the natural system in OFAT 8 is discussed in Section 4.2.6.

DOE indicated that the K_ds used in the analysis were representative of moderately oxidizing conditions, although the grout would support K_ds that reflected reducing conditions. NRC requested follow-up information on this response, as reductants (such as ground blast furnace slag) would need to be added to the grout to create reducing conditions, and NRC was not aware that DOE had committed to using reducing grout. In response to an NRC question on June 26, 2025, DOE stated that the "grout formulation will be developed during the closure planning process" (DOE-ID, 2025b). The NRC staff does not consider the moderately oxidizing K_ds to be conservative if the grout is not assumed to contain reductants.

In addition, the NRC staff recommends further consideration be given to the potential for limited mixing of waste with grout at the bottom of the bin sets, and the NRC staff plans to monitor the basis for the DOE assumption that the grout and waste will be well mixed (see Appendix A). If the waste were to form a discrete layer below the grout (which has not been ruled out by available information), radionuclide release may not be controlled by cementitious material partition coefficients, but rather by solubility. Higher dissolved radionuclide contents in water infiltrating through the waste could result, potentially leading to higher doses. This issue was previously raised by the NRC staff in the TER for the waste determination and PA for the INTEC Tank Farm Facility (NRC, 2006). The MFAT described in Table 4-4 partially addresses the technical issue with the chemical performance of the grout. In the MFAT, DOE assumes pessimistic K_d values for the grout and concrete base, and assumes the concrete base is fractured. DOE also assumes the lower of the grout or soil screening K_d value for the natural system. However, because DOE assumes that the SS bin sets still moderate flow through the engineered system in the MFAT (even though the flow rates increase earlier in time than for the base case; see Figure 4-8 in the Uncertainty Analysis Section), the peak dose results of the MFAT are lower than in other simulations (such as OFAT 1 and 2) that consider more extensive failure of the SS bin sets. However, OFATs 1 and 2 do not address the potential issue of flow occurring through preferential pathways bypassing the grout's chemical and hydraulic properties.

4.2.6 Far-Field Modeling

The overall flow and transport modeling approach for PA uses (i) an MCM model for the engineered barrier system, (ii) another MCM model for the vadose zone, and (iii) a GWSCREEN model for the SRPA aquifer. This section focuses on (ii) and (iii), or what is referred to as the far-field model.

Unsaturated zone hydrology and far-field transport is discussed in Section 3.2.2.7 and Section 4.1.1.2 of the PA (DOE-ID, 2022). DOE evaluates impacts to a hypothetical member of the public located at (i) the INL Site southern boundary, 12.7 km [7.9 mi] south-southwest of the CSSF, during the period of institutional control (2016 to 2116) and (ii) 100 m [330 ft] from the downgradient edge of each of the CSSF source locations during the compliance period (2117 to 3017) and post-compliance period (3017 to peak concentration).

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[21] It is unclear why the magnitude of the Pu-239 dose increased but occurred later in time in OFAT 5b compared to OFAT 5 [see Table 14 in DOE-ID (2025a)]. The Pu-239 K_d in cementitious materials was reduced to 50 L/kg (compared to 100 L/kg) and in sedimentary interbeds to 100 L/kg (compared to 1,140 L/kg).

The groundwater transport pathway considers radionuclide transport from each bin set to potential receptor wells located 100 m [330 ft] downgradient from each bin set. DOE modeled the groundwater transport pathway in three separate analyses: (i) source release from the engineered system, (ii) a vertical unsaturated zone transport stage extending from the base of each bin to the SRPA water table, and (iii) lateral transport in the SRPA from the area where the radionuclides enter the water table and are transported to the potential receptor well. Release from the engineered system in what is referred to as the "near field" model is evaluated in Section 4.2.5. This section focuses on the far-field model after the radioactivity is released into the natural system.

Vadose Zone Model Construction and Parameterization

Section 3.2.2.7 and Sections 4.1.4.1 and 4.1.4.2 of the PA (DOE-ID, 2022) discuss the unsaturated portion of the far-field model. The conceptual model is one-dimensional vertical flow through basalt and sedimentary interbeds that begins from the bottom of the concrete base and extends to the SRPA. The bin sets themselves are located in the alluvium (not modeled). The concrete base of each bin set is located above the uppermost basalt layer in a sequence of basalt and interbed sedimentary layers. The modeled total depth to the aquifer from the ground surface is 154.4 m [506.6 ft], and the alluvium thickness is 14 m [45.9 ft]. Subtracting the alluvium from the total unsaturated zone thickness gives a thickness of 140.4 m [460.6 ft] for the modeled unsaturated zone.

The 1-D vadose zone simulations implemented in the MCM model consider a unit gradient approach (no suction or pressure head from capillary forces), and the flow is equal to the unsaturated hydraulic conductivity provided the infiltration rate is less than the saturated hydraulic conductivity. Effective porosity is assumed to be equal to the total porosity. Moisture characteristic curves are used to determine the moisture content and unsaturated hydraulic conductivity. Physical dispersion can either be explicitly considered, or numerical dispersion can simulate the impact of physical dispersion. For more information on the MCM model see Section 4.1.1 of the CSSF PA (DOE-ID, 2022). Figure 4-2 of the PA shows the linkages between the MCM and GWSCREEN models.

The unsaturated zone was discretized into 63 cells consisting of either basalt or sedimentary interbeds [see Table 4-8 of the PA (DOE-ID, 2022)]. Based on the lithology provided by the Compilation of INTEC Geophysical Logs in the Vicinity of the Calcined Solids Storage Facility (Lawrence and Jolley, 2018), some basalt and interbeds were grouped together to reduce the total number of cells and provide relatively uniform cell thickness. DOE indicates that this simplification makes no difference in the overall simulation because the total basalt and interbed thicknesses were the same as those reported in Lawrence and Jolley (2018). Of the unsaturated thickness, 126.5 m [415 ft] was composed of basalt and 13.9 m [45.6 ft] was composed of interbed.

DOE-ID (2022) represented transport in the unsaturated zone with six vertical 1-D columns, each extending from the concrete base of a bin set to the water table. Each column has the same cross-sectional area as its bin set and incorporates seven sedimentary interbeds {13.9 m [45.6 ft]} total thickness] and eight layers of fractured basalt {126.5 m [415 ft] total thickness]. The vertical flow rate in the model is assumed to be elevated at 18 cm/yr [7.1 in/yr] for the first 50 years, decrease linearly with time to 1 cm/yr [0.4 in/yr] over the next 50 years, and remain at 1 cm/yr [0.4 in/yr] thereafter. With the assumed hydraulic parameters, the water velocity is ~130 times faster in the basalt fractures than in the interbeds with the long-term flow rate. The fine grains in the interbeds are assumed to attenuate releases from the bin sets through sorption, while sorption is assumed to be negligible in the basalt fractures, so that transport time is dominated by slow vertical flow in the interbeds with or without sorption.

DOE-ID (2022) cites previous work as showing extensive lateral redistribution associated with interbeds when large water sources are present (e.g., operational leaks, inflow from the BLR to drive lateral flow in perched zones) but expects that no such water sources to be present during the compliance period. Anthropogenic recharge is not expected after closure and BLR seepage is not expected near the bin sets. DOE-ID (2022) based the conceptual model of predominantly vertical 1-D flow in zones far from the BLR on a previous analysis (DOE-ID, 2011).

DOE-ID (2022) uses the Fortran-based MCM software (Rood 2021, 2010) to calculate 1-D flow and transport from the bin interior to the water table. The MCM model accounts for transient infiltration, radionuclide decay chains, sorption, and longitudinal dispersion. For each grid cell, (i) water mass draining to the grid cell over a time step is fully mixed with the cell contents, resulting in an updated cell average moisture content; (ii) a cell-average hydraulic conductivity is calculated using the cell-average moisture content and a material-specific moisture specific curve, and (iii) the flow rate of water exiting the grid cell is based on gravity drainage with the updated cell-average hydraulic conductivity. Transport calculations follow a similar approach, accounting for changes in sorbed radionuclide mass with changes in water content.

NRC staff considered the conceptual models, numerical approaches, and parameter values used to represent transport from the engineered system release point (the bottom of each CSSF concrete base) through the vadose zone to the SRPA water table and within the SRPA to the assumed receptor well locations. NRC staff performed several confirmatory calculations based on information provided in DOE-ID (2022).

To test implications of the gravity drainage assumption implemented in the MCM code, NRC used a model that integrated Darcy's Law from the release point to the water table using the same parameter values while including capillary forces. The interbeds tended to be wetter when capillary effects at the transition from interbed to fracture system were included, resulting in slightly slower transport through the interbeds. NRC staff concluded that calculated volumetric water contents are minimized under the assumption of gravity flow in a 1-D vertical flow model, resulting in lower advective travel times through the vadose zone.

Winfield (2003) measured properties for four interbed core samples from ICPP-SCI-V-214 and six from other nearby Vadose Zone Research Park wells, describing most samples as having a silty loam texture. Perkins (2003) measured properties for six interbed core samples from ICPP-SCI-V-215. The measured properties included volumetric water content at fixed rates of steady flow, corresponding to values of unsaturated hydraulic conductivity. NRC staff compared the relationship between hydraulic conductivity and volumetric water content with the parameters used by DOE-ID (2022) with these measured values. At ~1 cm/yr [~0.4 in/yr] flow, volumetric water content for the 16 core samples ranged between ~0.16 to ~0.46 (almost this entire range was found in ICPP-SCI-V-215) and the DOE-ID (2022) relationship predicted ~0.37. The DOE (2022) interbed hydraulic property values predict volumetric water content that are approximately in the middle of the spread of ICPP-SCI-V-214 volumetric water content values for fluxes between 1 and 10 cm/yr [0.4 and 4 in/yr]. Given the vertically variable nature of interbed properties and the low sensitivity of travel time to volumetric water content, NRC staff concluded that the relationship reasonably represents the range of observed interbed volumetric water content values.

Snake River Plain Aquifer or Saturated Zone Model Construction and Parameterization

Sections 3.2.2.8 and 4.1.4.3 of the PA (DOE-ID, 2022) provide details on the saturated zone model. DOE-ID (2022) represented regional flow in the eastern half of the SRPA as generally northeast to southwest along the axis of the Snake River Plain, with local variability due to inflows and aquifer heterogeneity. DOE-ID (2022) represented transport in the SRPA saturated zone near the bin sets as occurring in a unidirectional horizontal flow field from north to south. Flow is assumed to occur in the fractures of the basalt units in the saturated zone with an average linear velocity of 1 m/day [3.3 ft/day]. The model assumes that (i) recharge in the transport zone has a negligible influence on flow and transport, (ii) sorption is negligible in the basalt fractures (all radionuclides travel with the same velocity), and (iii) radionuclide ingrowth rates are negligible over the short time (3.3 months) between mass entering the aquifer and reaching the receptor wells. Each potential receptor location includes a well open over the top 15 m [49 ft] of the 76-m [249-ft]-thick SRPA.[22] Each receptor well is aligned to intercept the maximum concentrations emanating from its corresponding CSSF source area {i.e., 100 m [330 ft] from the edge of the source area and directly downstream of its centerline).

DOE-ID (2022) uses the aquifer portion of GWSCREEN Version 2.5a (Rood, 2003; software updated April 4, 2008) to calculate radionuclide transport in the saturated zone. The aquifer-transport model employs a 3-D semi-analytical solution to the advection dispersion equation for steady-state unidirectional flow in an aquifer of infinite lateral extent and finite thickness. The analytic solution is based on an instantaneous pulse of mass released from a rectangular source area at the top of a finite-thickness but laterally infinite aquifer with no initial radionuclide concentrations. Time-dependent mass flux to the aquifer is calculated based on release from the source zone with travel times through the vadose zone calculated based on the infiltration rate, calculated moisture content, and resulting unsaturated hydraulic conductivity. DOE then uses superposition to combine plumes from different CSSF sources at receptor locations and integrates the solution over the well screen to calculate the average concentration at any time. The source area is assumed to be square, with the same area as the CSSF bins. The approach does not account for pumping rates, which is equivalent to assuming that pumping generates negligible disturbance to the flow field (i.e., a negligible cone of depression around the borehole and no additional clean water is drawn into the well to dilute plume concentrations).

DOE-ID (2022) uses separate simulations for each bin set, combining the results from all simulations at each potential receptor well. Each simulation is provided with the radionuclide mass exiting from the vertical column, which is assumed to enter the top of the aquifer spread evenly over a square patch with the same area as the corresponding bin set. All mass arriving at the aquifer from the unsaturated zone is spread evenly over the patch area. Flow is assumed horizontal, with longitudinal, transverse horizontal, and transverse vertical dispersivities of 3.31, 0.662, and 0.00384 m [10.9, 2.17, and 0.0126 ft], respectively. The transverse horizontal and vertical dispersivities are 5 and 860 times smaller than the longitudinal dispersivity, resulting in essentially horizontal plumes that only slowly spread in the vertical direction.

All radionuclides are assumed to have zero K_ds in the basalt fractures of the SRPA. This assumption has been shown to provide upper-bound concentrations and greatly simplifies the calculations. The decay and ingrowth of radioactive progeny were treated explicitly in the unsaturated zone, and fluxes of parent and progeny were provided to the GWSCREEN aquifer model. However, further ingrowth of radioactive progeny after entering the aquifer was not

[22] The basis for the well screen of 15 m is from DOE-ID (1994, page C-11).

considered because transport times in the aquifer are much shorter than travel times through the unsaturated zone (a fraction of a year versus decades to centuries). Higher pore velocities and zero K_ds are the primary factors for relatively rapid aquifer transport times.

To test implications of the representation of transport in the SRPA, NRC reproduced the analytical model used in GWSCREEN. With the DOE assumed parameter values, changing the width of the source area along the flow direction or decreasing the width of the source area perpendicular to the flow direction had very little influence on peak concentrations at the receptor well, but spreading the plume vertically tended to decrease peak concentrations at the receptor well. NRC staff concluded that the assumed source area for transfer from the vadose zone to the saturated zone is reasonable for the GWSCREEN model with DOE-ID (2022) parameters.

With the assumed parameters, the analytical model calculates SRPA travel times that are <4 months for 100 m [330 ft] travel (i.e., from the source to the receptor well), consistent with minimal radionuclide decay before reaching the receptor well. The assumed parameters result in calculated plume concentrations that are rather narrow at the receptor well (concentrations 100 m [330 ft] downgradient from the source are much lower just a few tens of meters from the plume centerline) and tended to remain within a meter or two of the top of the aquifer. NRC concluded that a potential receptor well screened at the top of the aquifer would be expected to intercept the entire plume thickness.

The SRPA flow direction is somewhat uncertain, based on historical observations in the vicinity of the CSSFs that are disturbed by historical leakage. Because of the narrow plumes, the model would create little overlap in the plumes from multiple CSSFs unless the CSSFs are closely aligned with the flow direction. The selected direction is plausible and nearly aligns the two oldest CSSFs. In order to align other CSSF pairs (all of which would tend to be more robust than CSSF 1) the flow directions would be less consistent with regional flow models used by the DOE, so other flow directions would not be likely to generate greater concentrations at a receptor well. NRC staff concluded that the selected flow direction would be favorable for creating overlapping plumes, with other factors determining the amount of concentration increase.[23]

NRC concludes that the particular parameter values that DOE-ID (2022) selected for implementation in the MCM and GWSCREEN codes are reasonable to conservative given the conceptual models for flow.

It is unclear to NRC staff that the conceptual model for flow through the vadose zone is adequately supported, however. During operations, leaked radionuclides and ponding tests have penetrated the entire thickness of the vadose zone within years (DOE-ID, 2006). Parameters used in the MCM do not permit such rapid transit of the vadose zone. In response to Clarifying Comment (CC)-9, DOE-ID reports a travel time through the vadose zone of 37.4 and 555 years for a non-sorbing constituent at the assumed peak {18 cm/yr [7.1 in/yr]} and long-term {1 cm/yr [0.4 in/yr]} infiltration rates, respectively.

[23] In reality, the CSSF 1 plume does not appear to overlap the CSSF 2 and 3 plumes in time due to the timing of failure of the SS bins, and the CSSF 2 plume would be expected to be similar to the CSSF 3 plume. However, as discussed in Section 4.2.1 of this TER, the CSSF 3 peak dose appears to be an order of magnitude or more lower than the CSSF 2 plume.

The observed rapid transit of the vadose zone may be consistent with flow focusing processes that allow flow to concentrate within a fraction of the horizontal area, effectively bypassing part of the interbed by directing flow into “holes” (e.g., gaps in low-permeability layers within interbeds, thin zones in interbeds, especially transmissive through fractures, fractures that are not plugged with sediment). Flow focusing and lateral redistribution would be consistent with a wide range in travel times through the vadose zone (effectively a large dispersivity). Lateral flow within and above interbeds may be enhanced by bedding contrasts, heat-related alteration at the top of the interbeds as the basalt cooled, capillary contrasts between the interbeds and basalt fractures, and sloping basalt surfaces that create dipping and variably thick interbeds.

NRC recognizes that mass fluxes reaching the water table are sensitive to the release rate to the vadose zone, and travel time and dispersion in the vadose zone, with transport rates having a more significant effect on relatively short-lived radionuclides.[24] Radionuclide sorption can have a larger or smaller effect on point of exposure concentrations and dose dependent on the nature of the release (e.g., pulse-like releases versus nearly constant source terms). Minimal spatial spreading or mixing (e.g., due to diffusion, dispersion, or exchange of mass between faster and slower pathways) will lead to higher concentrations at the point of exposure. For a nearly constant source term, the release rate (e.g., as determined by the solubility or distribution coefficients in the waste zone or engineered system), will typically determine the concentration at the point of exposure with dispersion and natural attenuation mattering less. Section 4.1.4.1 of the PA (DOE-ID, 2022) notes that calculated mass rates crossing into the water table are insensitive to dispersion when the release rates to the vadose zone change slowly compared to travel time through the vadose zone and stated that mass fluxes were relatively constant for 1,000-year time steps with the radionuclides considered. Slow release or transport rates also led to substantial radionuclide decay for relatively short-lived radionuclides with the highest potential hazard and activity (i.e., Sr-90 and Cs-137).

The primary NRC technical issue with over-estimating travel time through the vadose zone is that calculations for relatively short-lived radionuclides may experience unrealistically large decay within the vadose zone, which is especially relevant for slow-release scenarios and for screening calculations. However, substantially higher release rates and dispersion would be needed to transport short-lived radionuclides to the point of exposure prior to radiological decay during transport, which is not likely given the expected robustness of the engineered system.

Transport Properties

The PA (DOE-ID, 2022) considers two types of media in the natural system: (i) fractured basalt and (ii) sedimentary interbeds to which it assigns its own set of hydraulic and chemical properties (e.g., sorption coefficients).

DOE-ID (2022) reuses parameters for the saturated zone from (i) a CERCLA PA using the same modeling approach (DOE-ID, 2011) and (ii) a regional groundwater model used to represent existing plumes and potential future plume evolution (DOE-ID, 2008).

The radionuclide transport model accounts for dispersion and equilibrium partitioning with the solid matrix, as well as moisture-content-related changes in storage and radionuclide decay. The PA (DOE-ID, 2022) uses a lower dispersion coefficient in the vadose zone to account for numerical dispersion inherent in the coarse grid discretization. DOE also estimated the

[24] Short-lived relative to the travel time to the point of exposure.

expected magnitude for physical dispersion (DOE-ID, 2022). DOE considered simulation results with the different dispersion coefficients, finding that the transport calculations were insensitive to longitudinal dispersion except for fast-decaying radionuclides, and selected the smaller implicit diffusion coefficient to improve computational speed.

The PA (DOE-ID, 2022) uses saturated zone dispersivity values based on (i) a longitudinal dispersivity value developed by DOE-ID (2003) from calibration to numerical simulations and (ii) ratios of transverse and vertical dispersivity to longitudinal dispersivity used by Whelan et al. (1996). Whelan et al. (1996) used the values to illustrate a numerical technique, citing Mills et al. (1985) for the dispersivity ratio values. Mills et al. (1985) provide an illustrative range of values from the then-current literature while discussing the mathematics of groundwater transport. Although not explicitly mentioned, the values cited by Whelan et al. (1996) are the extremes from the cited Mills et al. (1985) ranges that would result in the least spreading (i.e., highest peak concentrations).

Natural system sorption coefficients are provided in Table 4-12 of the PA (DOE-ID, 2022). No sorption was assumed for transport through the basalt.

NRC had comments regarding DOE's selection of sorption coefficients for the screening analysis and detailed analysis as expressed in RAI PA-4 (NRC, 2024). Table 4-8 lists distribution coefficients used in the CSSF PA versus Prikryl and Pickett (2007) recommended values. For example, some natural system K_ds utilized in the analysis were significantly higher than reasonably conservative values recommended in Prikryl and Pickett (2007). Furthermore, there was no apparent consideration of the potential transport impact in the vadose zone of leachate derived from the waste facility.

In response to NRC's RAI, DOE provided additional information to support its selection of distribution coefficients, as well as OFAT analyses to study the impact of uncertainty on the results. DOE indicated that the distribution coefficients used in the Phase II groundwater pathway screening analysis were based on Jenkins (2001), are consistent with values used in CERCLA assessments, and were previously approved by the EPA, the State of Idaho, and DOE. DOE also stated that, in general, the screening analysis adopted minimum values from Sheppard and Thibault (1990). NRC staff would note that the Sheppard and Thibault (1990) compilation is outdated. In a response to a 2008 NRC staff commentary on an error in Sheppard and Thibault (1990), Sheppard (2008) argued that site-specific data should be used when available, and that any default compilation used in the absence of site-specific data should be updated relative to the 1990 paper.

DOE stated that the INL (2011) K_d compilation used for the PA base case is more up to date than the Prikryl and Pickett (2007) reference cited in the NRC RAI. An inspection of data sources shows that many of the same key references were used in both reports. INL (2011) does cite some additional reports not in Prikryl and Pickett (2007)—particularly for uranium and strontium—but all but one were published before 2006. Another significant difference is that Prikryl and Pickett (2007) relied extensively on a K_d compilation in Appendix D of DOE-ID (2006) that focused on maximizing the use of site-specific data. The NRC staff notes that both compilations are likely outdated, being from roughly the same time period, and that their differences are likely related as much to the approach to synthesizing available data (e.g., in applying conservatism) as to the sources themselves. DOE indicated that cement leachate impacts in interbeds are ruled out in INL (2011) and DOE-ID (2018b), both prepared for the Remote-Handled LLW Facility PA. The rationale for excluding the effect in the interbeds was that, due to lateral transport of unaffected water that mixes with affected water, water

Table 4-8: Comparison of Distribution Coefficients (L/kg) Used in the Base Case Versus Recommended Values [from Prikryl and Pickett (2007)]

Radioelement	Material	INL CSSF PA	Prikryl and Pickett Recommended Values
Tc	Interbed	0.1	0
I	Interbed	3	0.1
Cs	Interbed	500	200
U	Interbed	10	1.6
Np	Interbed	18	5
Pu	Interbed	1,140	500

chemistry in the interbeds is dominated by water that has infiltrated without encountering cementitious materials. The CSSF facility is in a similar hydrologic setting with a similarly small footprint. The NRC staff finds this argument reasonable because, although the cement chemical impacts may be significant above the top interbed, the impacts will diminish in successively lower interbeds, such that sorption in interbeds below the top will be minimally impacted.

NRC staff also commented on the lack of consideration of uncertainty in the I-129 and Pu-239 K_ds. DOE indicated that the I-129 dose was too low to be a concern and that the Pu-239 inventory, while higher, had a K_d <1,000 L/kg, which was considered by DOE to be reasonably conservative, limiting its mobility and leading to the decay of Pu-239 prior to transport to the SRPA. Therefore, the potential impact of Pu-239 was stated to be primarily due to in-growth of its daughter U-235. Given the natural system K_d for U-235 of 10 L/kg, the potential risk from in-growth of U-235 from Pu-239 could lead to risk-significant results, although DOE notes that the K_d of the parent dictates the K_d of the daughter, leading to the peak dose from U-235 occurring after 100,000 years.

DOE did not adequately address NRC staff's comment about the uncertainty in the mobility of Pu, which may be associated with colloids or could be present in the more mobile Pu(VI) oxidation state. For the original screening analysis, DOE used the Pu interbed K_d from Jenkins (2001) of 22 L/kg. This value was originally proposed in the TRACK-2 Idaho National Engineering and Environmental Laboratory (INEEL) study for crushed basalt, which is expected to have a K_d at least 1 order of magnitude lower than that of interbed sediments. Some assumptions and observations of this K_d are included in the appendix in DOE/ID-10534 (Rodriguez et al., 1997). One key observation of Rodriguez et al. (1997) was that up to 50 percent of the INEEL Pu inventory could be mobile with a K_d of 22 L/kg (compared with a more realistic K_d value at least 10x larger for the remaining 50 percent) before Pu became risk-significant for groundwater. Experimental work at the time showed mobile fractions of around 1 percent; later work by Fjeld et al. (2001) found similar mobile fractions up to 2.4 percent. The Pu screening K_d appears conservative, although it still may not capture the impact of a small percentage of highly mobile oxidized Pu species (K_d<1). In some cases, K_d averaging will lead to an underestimate of the magnitude and overestimate of the timing of the mobile fraction dose, particularly if there is a significant difference in the K_ds.

To address uncertainty in the selection of K_ds, DOE ran new OFAT 8 with screening K_ds (including a value of 22 L/kg for Pu) instead of best-estimate K_ds for the natural system. DOE found that the results are insensitive to K_d for the natural system due to slow releases from the engineered system. The K_ds primarily affect timing but not the magnitude of peak dose due to the slow release of radionuclides from the engineered system. For a pulse release in the case of substantial underperformance of the engineered system, the magnitude of the peak dose can be sensitive to the K_d selected and explicit consideration of a more mobile fraction with lower K_d could lead to a higher dose dependent on the value and fraction of the inventory that is more mobile.

4.2.7 Dose Methodology

DOE screened out biotic pathways from detailed consideration in the PA due to depth of waste (see Sections 3.2.5, 3.3.3 and 4.4). NRC staff reviewed the information and thinks DOE provided sufficient information from the literature regarding depth of intrusion of various animals (e.g., harvest ants) or depth of plant roots (e.g., big sagebrush roots) to screen out the biotic pathway as a potential mechanism for exposure of humans to radioactivity at depth.

DOE performed detailed groundwater and air modeling (Sections 4.1, 4.2, 4.3 of the PA) to assess the dose to members of the public to demonstrate compliance with the performance objective in 10 CFR 61.41, "Protection of the general population from releases of radioactivity." The conceptual models for the air and groundwater transport pathways, and screening approach, are discussed in Section 3.2 of DOE-ID (2022). Exposure pathways and scenarios are described in Section 3.3. Only radionuclides that can partition in the gas phase were considered for the atmospheric pathway: H-3, C-14, and I-129. A radon flux screening analysis was also performed. Due to the depth of waste in the bin sets {more than 13.7 m [45 ft]}, radon is unable to migrate to the surface prior to decay. Only radon produced from waste in the transfer lines that will remain following closure (transfer lines greater than 3 m [10 ft] below ground surface—all shallower lines are assumed to have been removed) is considered in the radon analysis.

Gas phase transport of the three radionuclides considered was modeled using EPA's EMSOFT from the waste zone to ground surface. EPA's CAP-88 software was used to model air dispersion from the point source to the site boundary located approximately 18 km [11 mi] to the south-southwest.[25] The subsurface material above the waste zone was conservatively assumed to be alluvium, which has a higher permeability compared to undegraded grout, which will be used to stabilize waste in the bin sets.

The groundwater models used in the PA are discussed in preceding sections of this TER and are not repeated here. However, Section 4.5 of the PA describes the methodology used to calculate the annual all-pathways dose, which is based on the methodology in NRC (1977) and Peterson (1983). The pathways include (i) contaminated groundwater, (ii) leafy vegetables and produce irrigated with contaminated groundwater, (iii) milk and meat from animals that consumed contaminated water and contaminated fodder irrigated with contaminated groundwater. The receptors are assumed to be located 100 m [330 ft] down-gradient from each bin set.

RESRAD-ONSITE was used to calculate the Rn-222 flux at the surface and dose from inadvertent intrusion into the waste (see Section 4.3 of the PA). Radon precursors include Pu-242, U-238, U-234, Th-230, and Ra-226. RESRAD-ONSITE was also used to calculate dose to the inadvertent intruder (see Section 4.3). NRC staff found the use of the RESRAD-ONSITE code to be acceptable for use in calculating the potential impact from Rn-222, particularly given

[25] CAP-88 results from the nearby Remote-Handled Low-Level Waste Disposal Facility were actually used to estimate the doses based on pathway dose conversion factors (i.e., annual dose per unit release rate in Ci/yr).

the low risk-significance even when modeled using conservative approaches (the maximum peak radon flux from CSSF 1 due to a plugged transfer line within the 1,000 year compliance period was 1.6×10^{-3} Bq/m^2/s [4.3×10^{-2} pCi/m^2/s] compared to the performance standard of 0.74 Bq/m^2/s [20 pCi/m^2/s]). With respect to the inadvertent human intrusion calculations, DOE provided additional information to justify its selection of RESRAD-ONSITE biosphere parameter values, and NRC staff had no further questions (see Section 4.3 for additional details).

In response to the portion of NRC staff's RAI PA-6 related to biosphere modeling uncertainty (DOE-ID, 2025a), DOE indicated that the biosphere parameter uncertainty should not be evaluated in the PA uncertainty analysis. NRC staff agree that biosphere modeling assumptions regarding future human behavior do not have to be evaluated in an uncertainty analysis; however, DOE should use reasonably conservative behavioral and metabolic parameters or provide support for the values selected. NRC staff noted that the biosphere parameters were not consistently derived in the PA (i.e., the 10 CFR 61.41 and 10 CFR 61.42 used a different dose methodology) and that some of the biosphere parameters used in the PA are based on dated NRC references that could potentially underestimate the dose. NRC staff requested that DOE compare behavioral and metabolic values to those used in NUREG/CR-5512, Volume 3 (Beyeler et al., 1999), which are reported in NUREG-1757, Volume 2, Rev. 2, Table I-11 (NRC, 2022b). NRC staff also requested groundwater pathway dose conversion factors (PDCFs) to focus NRC staff's review.

NRC noted inconsistencies in the DOE justification for the biosphere parameters. The consumption rate comparisons made by DOE in Table 65 of the RAI responses (DOE-ID, 2025a) did not use the correct values from NUREG/CR-5512, Volume 3 (Beyeler et al., 1999), in the comparison.[26] Furthermore, DOE assumed that the fraction produced locally should be 0.25 percent, while no reduction in the NUREG/CR-5512, Volume 3, values should be made, because the data are based on home grown ingestion rates with 100 percent of the consumption rates assumed to be from local sources. The NRC comparison in Table 4-9 shows that the CSSF PA/CA significantly underestimates the consumption rates for leafy vegetables, which can have higher plant transfer factors leading to relatively higher doses. Table 4-9 also shows the CSSF PA/CA underestimates milk ingestion compared to the NUREG/CR-5512, Volume 3 values. While site-specific or regional specific values could be used to add more realism to the calculations, it does not appear that the generic values are conservative.

DOE also provided groundwater PDCFs in the PA (DOE-ID, 2022) and repeated those values in the RAI response (DOE-ID, 2025a). To increase the efficiency of its review, NRC staff compared these values to values used in other PAs for other sites. The values used in the INL CSSF PA were generally higher with the exception of the Np-237 PDCF (compared to the Yucca Mountain PDCFs). In general, DOE's biosphere parameters for the groundwater pathway appear to err on the side of higher doses for most of the radionuclides. DOE also notes in the RAI response (DOE-ID, 2025a) that the soil to plant transfer factors are an order of magnitude higher in NUREG/CR-5512, Volume 3 (Beyeler et al., 1999), compared to what was used in the INL CSSF PA based on Baes et al. (1984), which could partially explain the difference in the Np-237 PDCF. While the Np-237 dose could be higher than what DOE modeled in the CSSF PA, because the CSSF PA used a lower-than-expected PDCF, in most scenarios, the Np-237

[26] DOE relied on Table 6.20 in NUREG/CR-5512, Volume 3, to produce the values in the table, which were based on the Volume 1 values and were explicitly stated to be based on flawed assumptions and consequently updated in Volume 3.

Table 4-9: Comparison of DandD (NRC computer code) Biosphere Parameters to CSSF PA/CA Parameters

Food Category	DandD (kg/yr)	CSSF PA/CA (kg/yr)
Leafy	21.4	12.6
Other Vegetable, Fruit and Grain	112	123
Beef	40	38
Milk	233	45

dose was found to be insignificant and more than an order of magnitude higher PDCF would be needed for Np-237 to challenge the compliance limit. Therefore, the NRC staff determined that the lower PDCF for Np-237 in the CSSF PA does not impact the compliance demonstration for the large range of conditions DOE modeled in its PA and RAI responses.

NRC staff recommends that biosphere parameters be updated to reflect more modern sources of information to support future PAs at the site, particularly for site-specific and potentially risk-significant parameters such as distribution coefficients and transfer factors. However, NRC staff find the approach used by DOE to be reasonable.

4.2.8 Uncertainty Analysis

Uncertainty analysis methodology and results are presented in Chapter 6 of the PA (DOE-ID, 2022). DOE performed sensitivity analysis using OFAT analyses. The OFATs were evaluated to provide information about the impact of uncertainty on the assumed engineered barrier performance as well as natural system performance. CSSF PA Table 6-1 describes the OFATs and Table 6-2 provides the results (DOE-ID, 2022). See TER Section 4.2.2 for more information about the OFATs and the additional OFAT or MFAT analyses that were performed (DOE-ID, 2025a) in response to NRC's RAIs (NRC, 2024). Individual OFATs and MFATs are also described and evaluated by NRC in more detail in individual TER Sections 4.2.3 through 4.2.6, as applicable. DOE's use of OFATs and MFATs to address key uncertainties in the closed CSSF's ability to meet the 10 CFR 61.41 performance objective was instrumental in allowing NRC staff to have reasonable assurance that the performance objective could be met provided certain assumptions were verified during monitoring (see Section 4.5).

DOE also performed Monte Carlo analysis to evaluate the uncertainty in the CSSF PA results based on uncertainty in the parameter values. Uncertain parameter distributions are listed in Table 6-3 of the CSSF PA with results reported in Table 6-4 [concentrations] and Table 6-5 [all pathways effective dose] (DOE-ID, 2022). NRC staff thought that the uncertainty analysis did not capture the range of uncertainty in the results based on the uncertainty in the parameter values, as well as due to conceptual model uncertainties. NRC staff's RAI PA-6 (NRC, 2024) enumerated NRC staff's technical concerns including the following:

- Assumptions regarding engineered performance and particularly the use of a CDF of SS bin failure times that leads to lower, prolonged releases over tens to hundreds of thousands of years

- Lack of consideration of underperformance of multiple barriers

- Lack of consideration of higher infiltration rates {maximum infiltration rate of 2 cm [0.8 in]}

- Limited consideration of distribution coefficient and inventory uncertainty for only a limited number of radionuclides (Np-237, Se-79, and Tc-99) and range of values, although staff acknowledged that overly broad distributions on distribution coefficients could lead to risk dilution

- Lack of consideration of biosphere parameter uncertainty

In response to RAI PA-6, DOE described why its corrosion analysis was realistic and why uncertainty in SS barriers in OFATs and MFATs was sufficient, with new and old MFATs addressing key issues identified by NRC staff. DOE also provided justification on its consideration of uncertainty in infiltration rates based on uncertainty in precipitation rates and provided analyses in response to RAI PA-6 and CC-7 on why seasonal variations in infiltration are dampened at depth and approach a constant infiltration rate. DOE also justified its selection of radionuclides for consideration of uncertainty with respect to inventory and sorption coefficients—concluding that all other radionuclides besides Np-237, Se-79 and Tc-99 were insignificant with respect to potential dose. The uncertainty estimates are based on analytical or measurement uncertainty or assumed uncertainty for species that were undetected in liquid wastes or calcined waste such as I-129, Cs-135, and Am-243, and curium isotopes. While not rigorously derived, it was expected by DOE to account for uncertainty in waste processing campaigns and ORIGEN2 modeling.

It appeared to NRC staff that the uncertainty estimates only attempted to account for the expected decay between the first processing campaign in 1965 versus the assumed 1988 start date. Ratios are calculated with respect to easy-to-detect Cs-137 with an approximately 30-year half-life with the factor focused on the reduction in the inventory rather than on uncertainties that could lead to higher inventory. A factor of 2 lower and higher inventory was considered. DOE also noted that it expects the assumed volume of calcine to be pessimistic, and that the web-based Risk and Dose Calcine Assessment Tool (RADCAT), currently being developed (EDF-11612; DOE-ID, 2024), will allow DOE to re-calculate potential doses for the Base Case and alternative scenarios as additional data is collected on parameters such as inventory (e.g., RADCAT will be able to evaluate the impact of a higher or lower volume of calcine remaining in the bin sets following waste retrieval). DOE indicated that Tc-99 data are available in RPT-949 (Swenson, 2018) and are expected to be over-predicted by a factor of 2 based on comparison to measured data. DOE also indicated that I-129[27] and C-14 are not risk-significant (e.g., C-14 was screened out) and that I-129 is difficult to measure at the low concentrations present in the calcine due to its volatilization during the high heat calcination process. DOE indicated that based on the Base Case results, the inventory of Tc-99 would have to be two orders of magnitude higher, and the inventory of other radionuclides would need to be greater than two orders of magnitude higher, to challenge the 10 CFR 61.41 performance objective. While this is true based on the base case PA results, if alternative scenarios are considered, then less than an order of magnitude higher inventory could lead to doses close to the performance objective of 0.25 mSv/yr. DOE characterizes these scenarios as highly unlikely, however.

DOE presents results of sensitivity analyses that show the lack of sensitivity of the magnitude of peak dose to distribution coefficient, which affects the timing but not the magnitude of peak dose. NRC staff note that this is primarily due to the assumed conceptual model for flow through the system based on the area of the bin sets breached due to corrosion (and the very low

[27] DOE also notes that I-129 is assumed to be 1 percent of the original activity because less than 1 percent of the iodine went to calcine (a small percentage went to the High-Level Waste Evaporator).

corrosion rates), which serves to prolong the releases over long time periods. As discussed in Section 4.2.5, if more discrete failure is assumed, then the significance of the distribution coefficient could be much greater.

With respect to the potential for risk dilution due to overly broad parameter distributions, DOE evaluated deterministic versus probabilistic results and peak of the mean versus mean of the peak doses, which were within a factor of 2, and concluded that there was no risk dilution. In fact, overly broad distributions affecting the timing of peak dose could lead to lower peak of the mean doses (compared to mean of the peaks dose), as discussed in NUREG-1757, Volume 2, Rev. 2 (NRC, 2022b). This type of risk dilution could occur due to parameters such as K_d that affect the timing of peak doses due to the spreading out in time of peak dose. However, NRC staff was more focused on the prolonged ramping up of flow through the waste zone, which served to prolong releases that occurred over tens to hundreds of thousands of years and thereby limited the peak dose. In contrast, if DOE modeled discrete failure of the bin sets, similar to other DOE PAs (e.g., SRS tank farms), the peak doses could be considerably higher. Of course, use of metrics such as peak of the mean when more discrete failures are modeled, may underestimate the peak dose, if the discrete failures are spread out over large time periods (e.g., the peak doses could be relatively high but could be averaged with other realizations with different times of failure thereby leading to significantly lower mean doses over time and consequently lower peak of the mean dose). In the case of discrete failures, a comparison of the mean of the peak to the peak of the mean could reveal this phenomenon. While DOE's evaluation of the peak of the mean versus the mean of the peak is helpful in showing the current probabilistic analysis does not exhibit the phenomena of risk dilution, it does not address the key issue regarding the uncertainty in the conceptual model for waste release (i.e., that the percent area of the SS area breached dictates the amount of flow through the system and that the SS bins will take over a million years to completely fail leading to very slow, and prolonged releases of radioactivity into the environment). See for example, Figures 4-6 and 4-7 for CSSF 1 and 2 failure times used in the probabilistic analysis, respectively; and Figure 4-8 which shows the CDF used in the base case and OFAT 7 (stainless steel in contact with calcine).

Additionally, NRC staff indicated that alternative conceptualizations involving common cause failures may be more realistic than assumed in the PA and could lead to more rapid failure affecting multiple barriers (e.g., lack of mixing of waste with grout coupled with lower retention of waste in grout; preferential flow of water along steel bin walls and through the waste zone leading to enhanced corrosion and higher release rates; earlier steel bin failure; and greater infiltration rates through the engineered system). Therefore, in response to RAI PA-6, DOE included an additional MFAT to address NRC comments regarding multiple barrier failure. DOE indicated that this barrier analysis considered pessimistic assumptions concerning the (1) lifetime of the SS bins, and (2) lifetime of the grout, (3) K_ds in cementitious materials and natural system, and (4) natural infiltration rates. The CDF used for the MFAT analysis was based on the OFAT 7 analysis with higher corrosion rates for SS in contact with calcine. The peak dose was a factor of 58 higher compared to the peak dose from the Base Case (0.11 mSv/yr versus 1.2×10^{-3} mSv/yr); however, the doses were still not as high as OFAT 1c with notable differences being even faster corrosion and fracture flow through the system, as well as higher infiltration rates and more pessimistic sorption coefficients. The MFAT did show more significant doses from Np-237, Pu-239, Pu-240, and Pu-242 (see Table 68 and Figure 86 in the PA). Therefore, NRC staff determined that uncertainty in the sorption coefficients for Pu should have also been included in the MFAT analysis considering that Pu exists in multiple oxidation states, and because experiments suggest a small percentage of highly mobile Pu species may be present in the oxidizing INL environment.

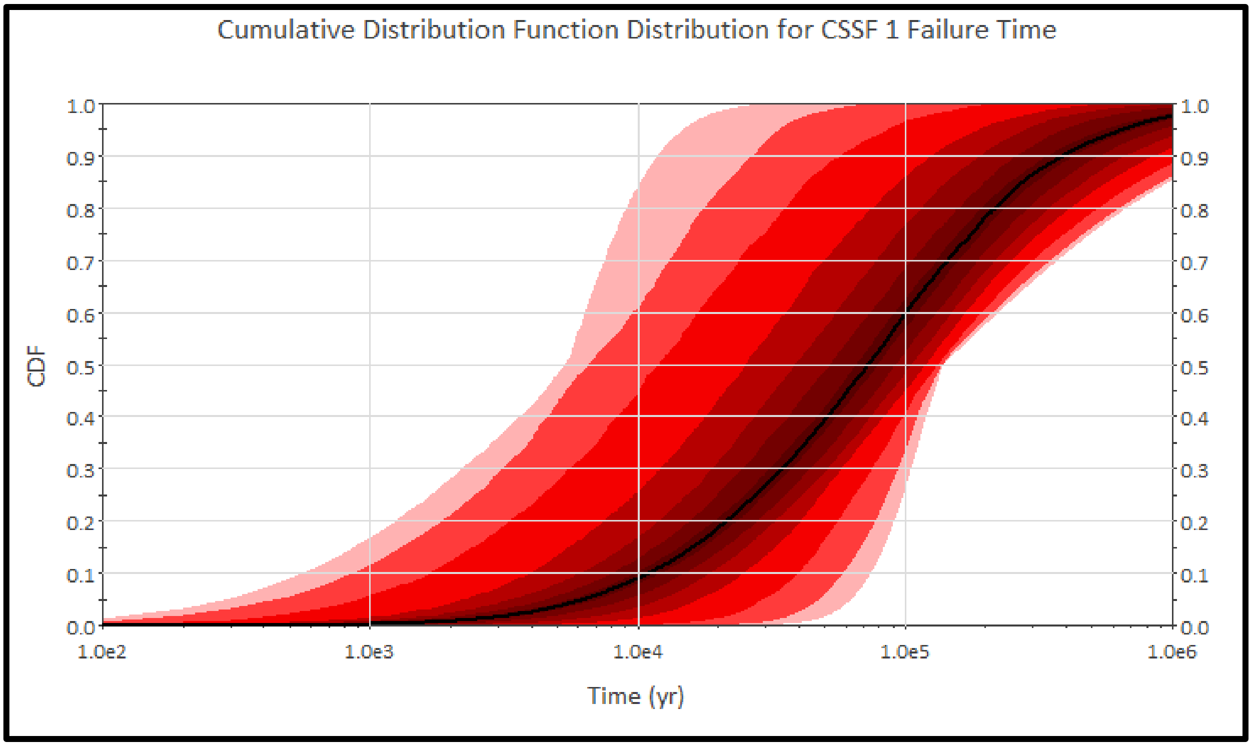

Figure 4-6: Uncertain CDFs for CSSF 1 in the INL INTEC CSSF PA (truncated at 1×10^6 years)

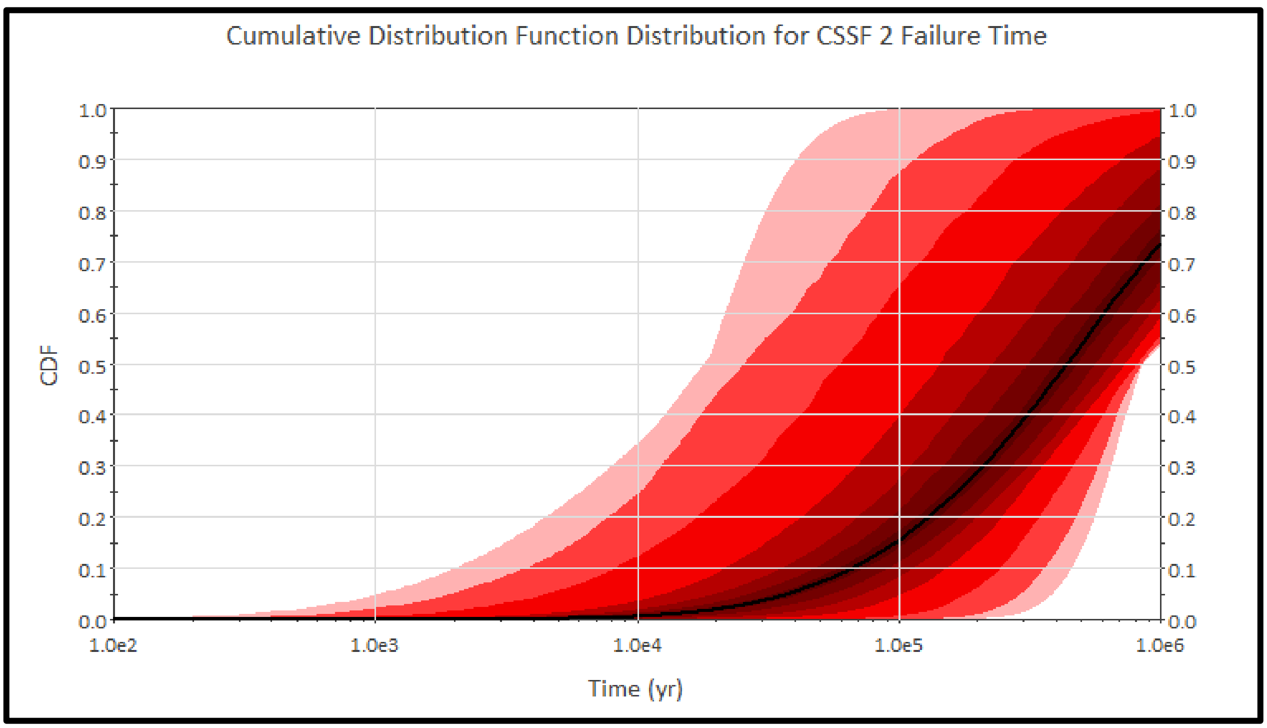

Figure 4-7: Uncertain CDFs for CSSF 2 in the INL INTEC CSSF PA (truncated at 1×10^6 years)

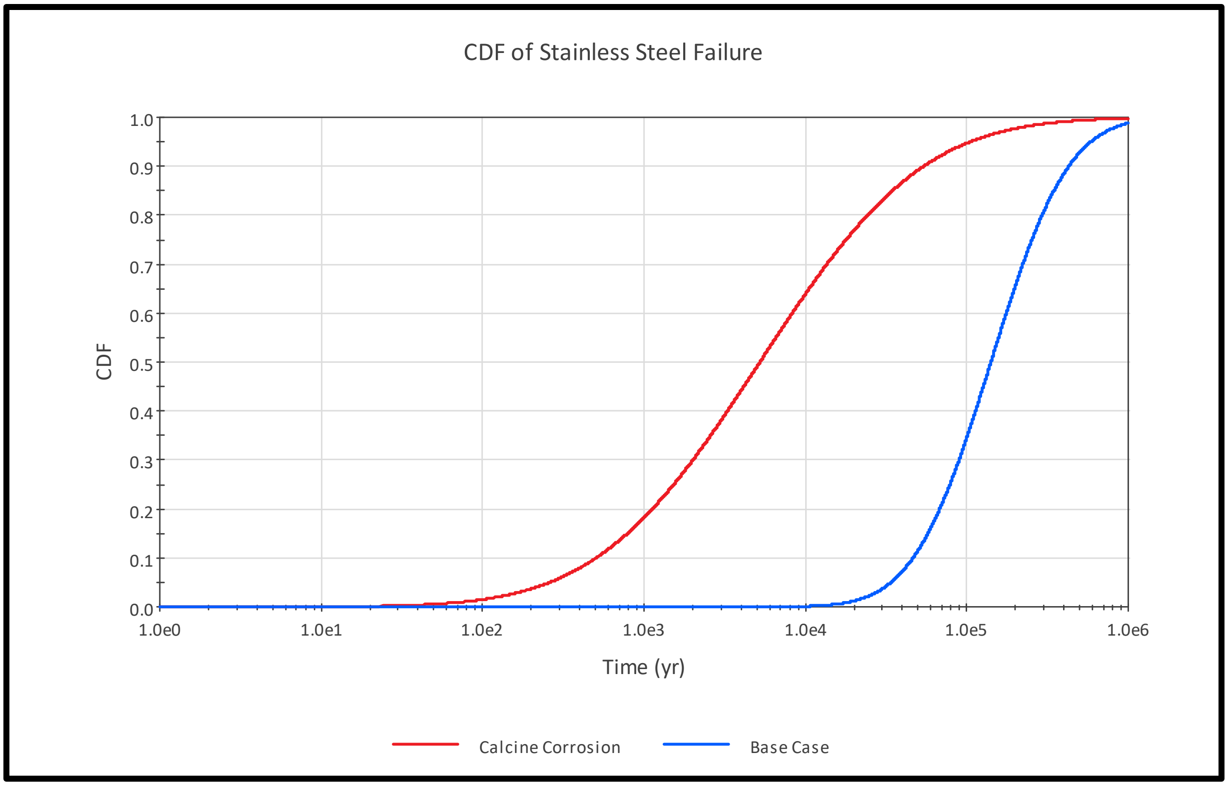

Figure 4-8: CDF of Updated CSSF 1 Analysis Using Corrosion Rates in Contact with Calcine Compared to Base Case Corrosion Rates Based on Passive Soil Conditions

While DOE did not perform an updated probabilistic analysis, DOE performed a series of additional OFAT and MFAT analyses addressing potential technical issues enumerated in NRC staff's PA RAI-6. The results of the additional analyses were all less than 0.25 mSv/yr. Therefore, NRC staff have reasonable assurance that the 10 CFR 61.41 performance objective can be met provided key assumptions in DOE's PA are met.

4.2.9 Conclusions Regarding Protection of the Public

NRC staff assessed DOE's ability to comply with the 10 CFR 61.41 performance objective by reviewing DOE's PA (DOE-ID, 2022), responses to requests for additional information (DOE-ID, 2025a,b) and supporting references. DOE generally followed NUREG-1854 (NRC, 2007) review procedures in the areas of infiltration and erosion control; near-field modeling, including engineered barrier degradation and waste release; far-field modeling; dose assessment methodology; and uncertainty/sensitivity analysis to demonstrate compliance with the 10 CFR 61.41 performance objective "Protection of Members of the Public from Releases of Radioactivity." DOE provided sufficient information to support the conceptual and mathematical models comprising the integrated PA, the data sources used to support the PA models, and evaluated uncertainty in the parameters, models, and results. DOE considered various features, events and processes in evaluating the impact of alternative conceptual models and exposure scenarios on the results. Therefore, NRC staff has reasonable assurance that the 10 CFR 61.41 performance objective can be met for the CSSF facility if key assumptions identified in the monitoring areas are verified (see Appendix A). NRC staff will monitor DOE's closure plans and activities as they progress, including review of grout formulations and studies to provide support for key assumptions in the PA to assess DOE compliance with the 10 CFR 61.41 performance objective.

4.3 Protection of Intruders

The inadvertent intruder calculations are presented in Chapter 7 of the PA (DOE-ID, 2022). Various scenarios were considered that could bring waste to the surface where a member of the public could be exposed. DOE ruled out the basement excavation scenario because the minimum depth to waste in the bin sets is 13.7 m [45 ft] and the transport lines located above the ground surface down to 3 m [10 ft] below the ground surface will be removed as part of closure activities. Basement excavation below 3 m [10 ft] is, therefore, not assumed to occur given the depth of potential waste.

DOE assumed that intrusion does not occur until after 500 years following closure, because the bin sets are greater than 13.7 m [45 ft] below ground surface and protected by reinforced concrete vaults. DOE also assumes that the transfer lines are not subject to inadvertent intrusion until 500 years, because all transport lines are contained within a larger protective piping encased in reinforced concrete.

DOE considered the following exposure scenarios:

- Drilling (acute exposure) any time after 500 years post-closure of the bin sets and transport lines.

- Post-drilling agricultural (chronic exposure) any time after 500 years post-closure for the bin sets and transport lines.

A well is assumed to be drilled 122 m [400 ft] to the SRPA. A 22-inch well diameter is assumed for the acute exposure scenario, and a 20 cm [8 in] well diameter is assumed for the chronic exposure scenario. The drill cuttings are assumed to be distributed over 2,200 m^2 [24,000 ft^2] even though the total volume of drill cuttings would be spread over only 12.7 cm [5 in]. See Figure 4-9 for an illustration of the acute intruder exposure scenario. RESRAD-ONSITE is used to calculate the dose to the inadvertent intruder after the drill cuttings are assumed to be distributed over the surface.

The results of the inadvertent intruder analyses showed that doses are dominated by Cs-137 and Sr-90 even after 500 years, because these radionuclides have the largest inventories (1.3×10^{15} and 1.2×10^{15} Bq [3.6×10^{4} and 3.2×10^{4} Ci], respectively). However, given their relatively short half-life, the risk from Cs-137 and Sr-90 drops off significantly after a few hundred years (see Figure 4-10 for the results for the acute well driller scenario which are higher compared to the chronic exposure scenario.)

NRC noted in RAI PA-5 (NRC, 2024) that a key assumption in the intruder assessment is that all ancillary equipment within 3 m [10 ft] of the ground surface will be removed. This assumption eliminates consideration of a basement excavation scenario in the intruder analysis, which is significant because the excavation scenario leads to substantially higher potential intruder doses than a drilling scenario due to the larger quantity of radioactivity brought to the surface in the basement excavation scenario. Because transport lines are located closer to the surface and have less robust intruder protection compared to waste located in the bin sets, the transport lines may be more susceptible to human activity including construction projects that could lead to disturbance of radioactivity which could be brought to the surface, where a member of the

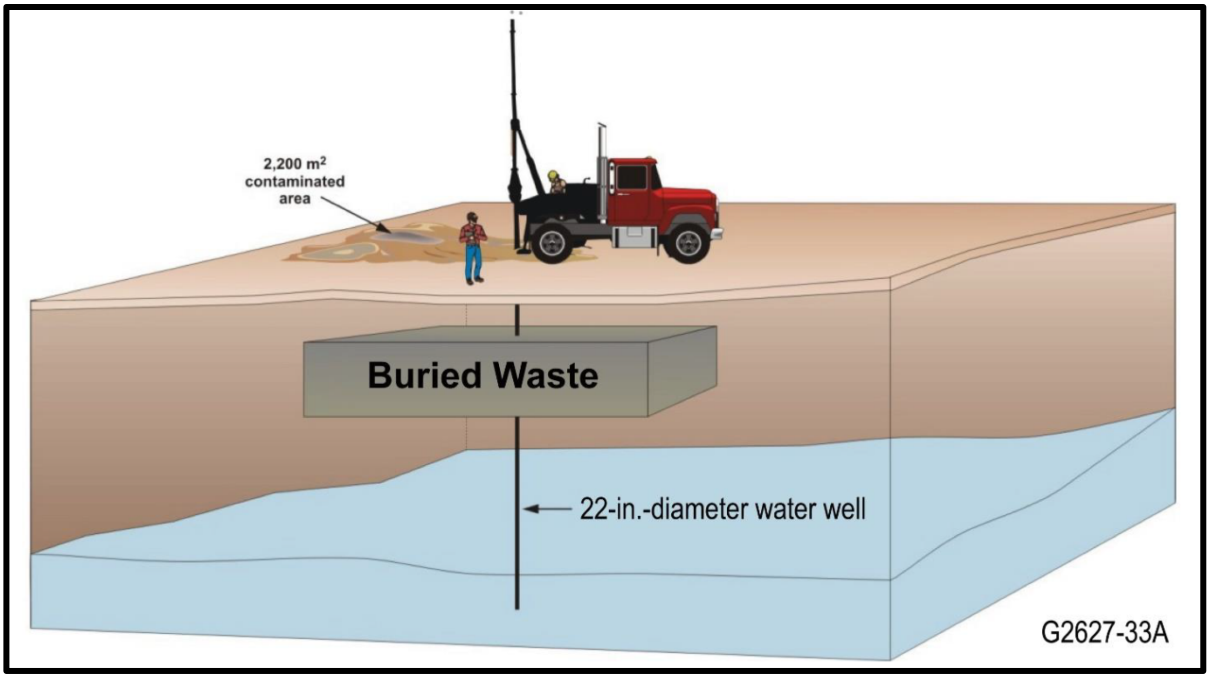

Figure 4-9: **Acute Intruder Drilling Scenario. Image Credit: Figure 7-1 from the INL CSSF PA (DOE-ID, 2022).**

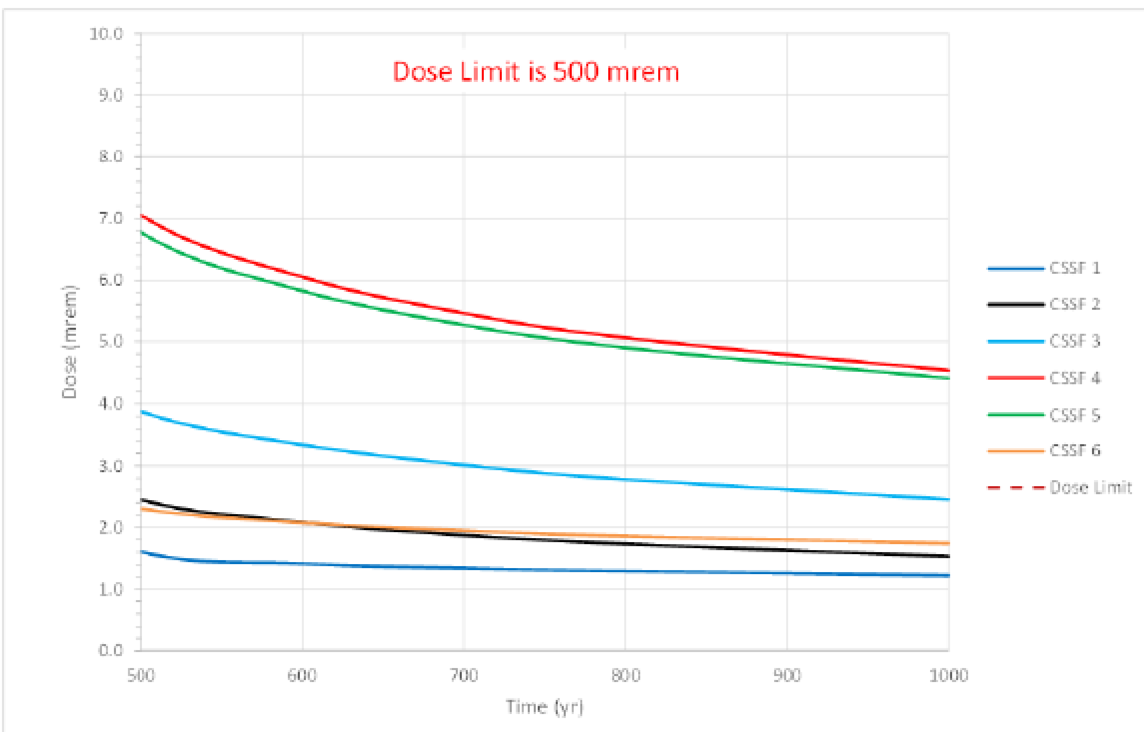

Figure 4-10: **Results of the Acute Inadvertent Intruder Calculation from the INL INTEC CSSF PA (DOE-ID, 2022). Image Credit: Figure 7-5 in the INL INTEC CSSF PA.**

public could be exposed. Additional information regarding planned closure activities (e.g., binding commitments to remove all impacted infrastructure within 3 m [10 ft] or more of the ground surface), controls, and barriers to intrusion to prevent excavation into the transport lines following closure of the bin sets was requested by NRC staff to support elimination of a more risk-significant excavation scenario from consideration in the PA. To address NRC staff's comment, DOE provided additional information regarding CSSF Tier I & II closure plans required by DOE M 435.1-1, which will include commitments for removal of transport lines and soil contamination within 3 m [10 ft]. NRC staff will monitor DOE's closure plans as they progress to ensure that commitments are made to remove ancillary equipment within 3 m [10 ft] of ground surface.

NRC staff also requested additional information regarding the potential for accumulation of waste in the transport lines, and clarify the basis for the potential accumulations depicted in Figure 2-42 of the PA (DOE-ID, 2022) and Figure 2-43 in the waste determination (DOE, 2023) reproduced below in Figure 4-11 (e.g., clarify if the accumulations were based on characterization data or process information alone).[28] NRC staff also questioned whether DOE had plans to characterize transport lines in the future to validate the assumed accumulation volumes and residual film volumes remaining in the transport lines.[29] In response to RAI PA-5 (DOE-ID, 2025a), DOE indicated that EDF-11119 (DOE-ID, 2018a) provides the most recent compilation of the current knowledge about the state of residual contamination in the transport lines. The air transport system sent cold material through the system and in other instances when operations switched to the next CSSF no cold material was sent through the system (e.g., between filling CSSF 2 and 3). Deposits developed in dead legs or in transport lines no longer in use (e.g., CSSF 2 solids transport line, the stub outs of CSSF 3 solids transport line, and possibly in the CSSF 4 transport lines when operations switched between 4 and 5). The rest of the transport lines were assumed to have a film of calcine residue 0.35 mm [0.014 in] on internal surfaces. There are currently no plans to sample the residual calcine in the transport lines in the future.

The inventory for the transport lines was based on an assumed volume of waste from the associated bin that might remain in the transport lines that were used in the intruder calculations. DOE assumed drilling through a 7.6 cm [3in] transport line, which is assumed to be one-twenty-fifth full of waste (i.e., 3.9 percent of the line volume filled with residual waste) or a 0.75 mm [0.030 in] film. A 56-cm [22-in] diameter drill bit would bring 9.95×10^{-5} m^3 [3.51×10^{-3} ft^3] of contaminated drill cuttings to the surface and a 20-cm [8-in] diameter well would bring 3.56×10^{-5} m^3 [1.26×10^{-3} ft^3] to the surface. These cuttings are assumed to be entirely composed of waste. DOE provided additional information on the basis for piping inventory in response to an additional information request (DOE-ID, 2025b) from the June 26, 2025, public RAI response meeting (see Table 4-10).

[28] EDF-11119 (DOE-ID, 2018) indicates that the air transport system operated in such a way that plugs developed in dead space of the transport lines, such as dead legs or solids transport lines no longer in use. For example, potential deposits or plugs to CSSF 2 and 3 are likely hot material because processing operations switched to filling CSSF 2 and 3 without using cold material. It is important to note that the waste determination (DOE-ID-2022-01) makes a point to refer concentration of calcine waste in transport lines as "deposits" or "accumulations" rather than "plugs" due to the high velocity air used to prevent solids from falling out or salting.

[29] Archibald and Demmer (1995) estimates remaining volumes for the waste calcine facility based on an assumed coating of waste of 0.35 mm [0.014 in] on internal surfaces which is extrapolated in DOE-ID-2022-01 to the transport lines located outside of the waste calcine facility for use in the CSSF PA.

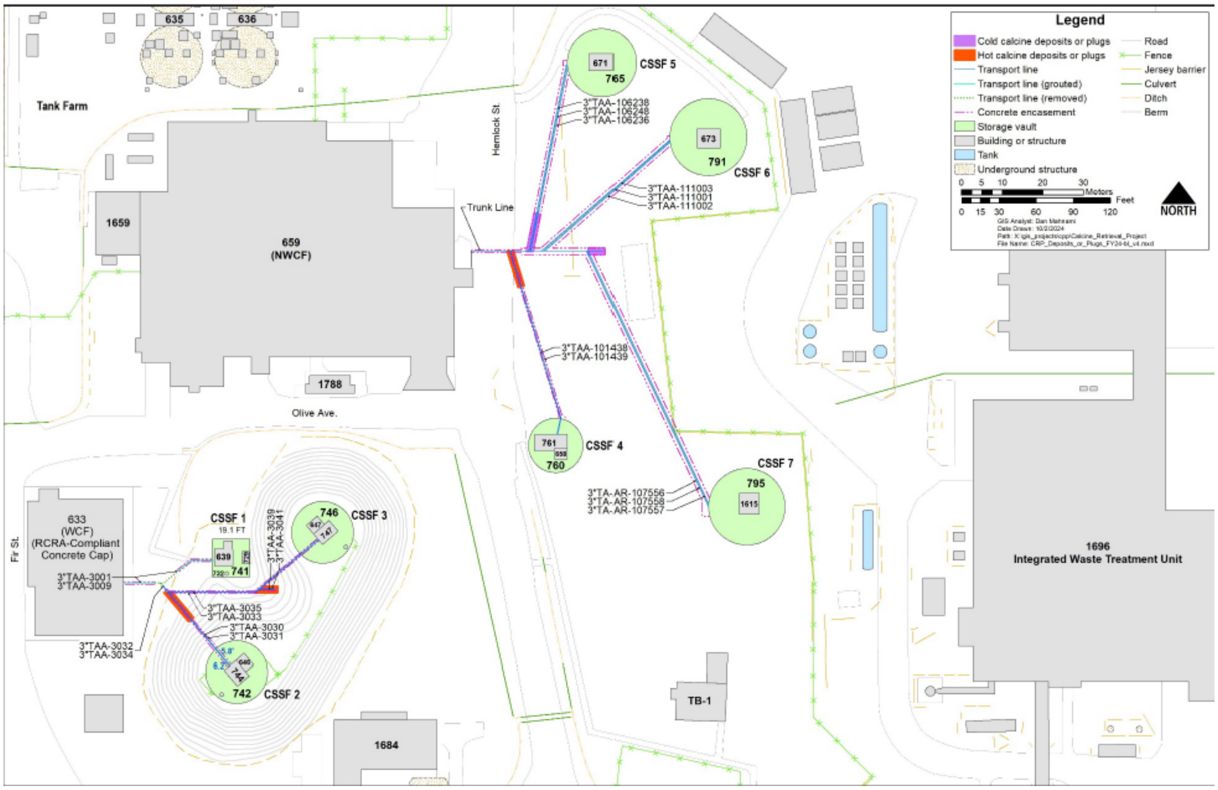

Figure 4-11: Revised Map of Figure 2-43 in DOE-ID (2022) and Figure 2-44 in DOE (2023). Image Credit: Figure 98 in DOE-ID (2025a).

NRC also noted in RAI PA-5 (NRC, 2024) that relatively short-lived radionuclides such as Sr-90 and Cs-137 have the greatest initial activities but their concentrations decay approximately an order of magnitude every 100 years and therefore, the doses from these radionuclides could be much more significant if receptors are exposed to the CSSF inventory within a few hundred years following closure, although unlikely given control of the site by DOE which is expected into the foreseeable future. DOE provided additional sensitivity analysis on the timing of intrusion into the transport lines starting at 100 years after closure, which is expected to be a very low likelihood event given DOE control of the site. The results for the chronic intruder drilling scenario for the CSSF 5 transport lines by pathway and time are presented in Table 23 and Figures 25 and 26; and key radionuclides (Am-241, Cs-137, Nb-94, Pu-238, Sr-90, and Tc-99) are presented in Figure 27 of DOE-ID (2025a). The doses were orders of magnitude higher at 100 years compared to 500 years, which is expected given the dominance of relatively short-lived radionuclides Sr-90 and Cs-137 to dose with almost all of the dose coming from the external gamma (Cs-137) and plant ingestion (Sr-90) pathways. In later years, long-lived Tc-99 dominates the much lower dose due to plant ingestion. The highest results for the acute intruder drilling scenario for accumulations in the CSSF 4 transport lines by pathway and time are presented in Table 21 and Figures 22 and 23; and key radionuclides (Am-241, Cs-137, Nb-94, Pu-238, Pu-239, Pu-240, and Sr-90) are presented in Figure 24 of DOE-ID (2025a). The results are significantly higher at 100 years although not quite as disparate as the chronic intruder results with Cs-137 driving dose through the external gamma pathway at earlier times and Am-241 and Pu isotopes driving the inhalation dose at later times albeit at much lower levels.

Table 4-10: Select Radionuclide Inventories Used to Represent Accumulations in the Transport Lines for the Intruder Calculations. Modified Based on Table 1 in the DOE Response to NRC's Follow-up Questions in the July 26, 2025, Public Meeting (DOE-ID, 2025b).

Radionuclide	CSSF 2 (Ci)	CSSF 3 (Ci)	CSSF 4 (Ci)	CSSF 5 (Ci)	CSSF 6 (Ci)
Am-241	1.33×10^{-4}	2.15×10^{-4}	3.14×10^{-4}	2.84×10^{-4}	5.59×10^{-5}
C-14	1.05×10^{-10}	6.92×10^{-14}	9.27×10^{-14}	9.62×10^{-14}	3.65×10^{-14}
Cs-137	1.97×10^{-1}	1.63×10^{-1}	2.19×10^{-1}	2.28×10^{-1}	8.64×10^{-2}
I-129	1.46×10^{-9}	1.08×10^{-9}	1.44×10^{-9}	1.52×10^{-9}	5.73×10^{-10}
Nb-94	1.93×10^{-6}	4.33×10^{-6}	5.80×10^{-6}	6.02×10^{-6}	2.29×10^{-6}
Np-237	2.07×10^{-7}	7.12×10^{-7}	3.99×10^{-6}	3.70×10^{-6}	7.67×10^{-7}
Pu-238	9.50×10^{-4}	1.48×10^{-3}	3.37×10^{-3}	3.19×10^{-3}	6.97×10^{-4}
Pu-239	2.13×10^{-5}	3.96×10^{-5}	9.43×10^{-5}	8.85×10^{-5}	4.66×10^{-5}
Pu-240	1.68×10^{-5}	2.86×10^{-5}	6.07×10^{-5}	6.25×10^{-5}	2.51×10^{-5}
Pu-241	4.76×10^{-4}	7.60×10^{-4}	1.62×10^{-3}	1.69×10^{-3}	6.85×10^{-4}
Se-79	5.88×10^{-7}	4.45×10^{-7}	5.94×10^{-7}	6.21×10^{-7}	2.33×10^{-7}
Sn-126	2.37×10^{-6}	1.79×10^{-6}	2.39×10^{-6}	2.50×10^{-6}	9.43×10^{-7}
Sr-90	1.74×10^{-1}	1.40×10^{-1}	2.03×10^{-1}	2.11×10^{-1}	7.46×10^{-2}
Tc-99	8.99×10^{-5}	6.58×10^{-5}	8.76×10^{-5}	9.22×10^{-5}	3.48×10^{-5}
Th-230	1.43×10^{-8}	5.45×10^{-10}	2.40×10^{-10}	7.12×10^{-9}	4.04×10^{-9}
Th-232	7.79×10^{-15}	3.05×10^{-15}	4.09×10^{-15}	4.25×10^{-15}	1.61×10^{-15}
U-232	1.03×10^{-9}	8.70×10^{-9}	1.41×10^{-8}	9.59×10^{-9}	1.88×10^{-9}
U-233	2.78×10^{-11}	1.12×10^{-10}	6.39×10^{-10}	5.29×10^{-10}	1.01×10^{-10}
U-234	7.84×10^{-7}	1.75×10^{-7}	3.70×10^{-7}	6.59×10^{-7}	4.13×10^{-7}
U-235	4.68×10^{-9}	1.73×10^{-9}	3.28×10^{-9}	8.53×10^{-9}	1.01×10^{-8}
U-236	1.18×10^{-8}	4.60×10^{-9}	8.96×10^{-9}	2.64×10^{-8}	2.27×10^{-8}
U-238	2.66×10^{-10}	3.05×10^{-10}	1.77×10^{-9}	4.66×10^{-9}	8.01×10^{-9}
Zr-93	3.99×10^{-5}	4.54×10^{-5}	6.09×10^{-5}	6.32×10^{-5}	2.40×10^{-5}
1 Ci = 3.7×10^{10} Bq					

NRC review of a subset of the biosphere parameters supporting the intruder assessment revealed a potential for underestimate of dose given the reliance on dated and generic references and use of RESRAD-ONSITE default parameter values. See NRC RAI PA-5 for additional information on NRC staff's technical issues with the biosphere modeling (NRC, 2024). DOE performed a sensitivity analysis on biosphere parameters to provide support that the 10 CFR 61.42 performance objective could be met with use of more modern sources for biosphere parameter values. Parameter values from Table I.11 of NUREG-1757, Volume 2, Rev. 2 (NRC, 2022b) were used in the sensitivity analysis as these behavioral and metabolic values were approved for use by NRC for decommissioning. The results of the analyses showed that the inhalation pathway drove the dose for the acute exposure scenario for CSSF 4 bin set from Am-241 and Pu isotopes, while the plant ingestion pathway drove the dose for the chronic pathway primarily from Tc-99 and Sn-126 for CSSF 1 bin set. NRC would note that the plant ingestion pathway appeared to be artificially reduced by 50 percent, which is the default assumption in RESRAD-ONSITE for any exposure areas greater than 1,000 m^2 [11,000 ft^2]. However, the plant consumption rates should be used without a 50 percent reduction (i.e., the consumption rates reflect home grown produce and a contaminated fraction of “1” should have been used with those rates). Nonetheless, even if the plant ingestion doses were doubled, the results of the sensitivity analysis showed that the doses were significantly less than 5 mSv/yr.

Finally, NRC staff noted in RAI PA-5 (NRC, 2024) that the intruder analysis considers the distribution of drill cuttings over a 2,200 m^2 [24,000 ft^2] area, leading to a thickness of contamination of 1.3 cm [0.5 in] for the acute exposure scenario and over a 2,200 m^2 [24,000 ft^2] area, leading to a thickness of contamination of 0.18 cm [0.07 in], which is then assumed to be tilled into 30.5 cm [12 in] of soil in the chronic exposure scenario.

Therefore, the assumed area may lead to an underestimate of dose for certain radionuclides and pathways. Distribution of drill cuttings over such a minimal thickness is not considered realistic. NRC recommended DOE perform sensitivity analysis on the area and thickness of contamination to ensure the doses are not underestimated. The 2,200 m^2 [24,000 ft^2] distribution area was adjusted to a 1,100 m^2 [12,000 ft^2] distribution area (and the thickness increased by a factor of two). The results of the sensitivity analysis showed that the doses could go up by a factor of 2 for the chronic exposure scenario for the plant ingestion pathway given the larger thickness of contamination. In all cases, the doses were less than 5 mSv/yr [500 mrem/yr], which is the dose described in NUREG-0782 (NRC 1981, Volume 2, Section 4.5) used by NRC to evaluate the 10 CFR 61.42 performance objective.

DOE provided additional information and performed extensive sensitivity analysis in response to staff’s RAIs to provide NRC staff confidence that the 10 CFR 61.42 performance objective could be met. In all cases, although the doses could increase above the Base Case values, the doses were significantly below the 5 mSv/yr dose benchmark used to evaluate the inadvertent intruder calculations. Therefore, NRC staff has reasonable assurance that the 10 CFR 61.42 performance objective can be met. NRC staff will monitor DOE’s closure plans and activities as they progress to assess DOE compliance with the 10 CFR 61.42 performance objective.

4.4 Other Performance Objective Results

4.4.1 Protection of Individuals During Operations (10 CFR 61.43)

To demonstrate compliance with the 10 CFR 61.43 performance objective, DOE provided information about its radiation protection program that helps ensure protection of workers and members of the public during facility operations. DOE requirements were stated to be comparable to the relevant requirements in the 10 CFR 61.43 performance objective. DOE also indicated that its regulatory and contractual requirements establish dose limits based on 10 CFR Part 835, “Occupational Radiation Protection,” and relevant DOE orders (e.g., DOE O 458.1, Change 4, “Radiation Protection of the Public and the Environment”). The dose limits correspond to the 10 CFR Part 20 radiation protection standards that are referenced in 10 CFR 61.43.

DOE indicates that its radiation protection program during operations and closure of the CSSF are consistent with as low as reasonably achievable principles as implemented by the ICP Radiation Protection Program.

DOE also cross-walks each of the 10 CFR Part 20 radiation protection standards listed in NUREG-1854 (NRC, 2007) to relevant DOE regulations and orders in Table 7-12 of the CSSF waste determination (DOE, 2023) and provides supporting text to show how the DOE regulations and orders are comparable to each of the 10 CFR Part 20 requirements. Further, measures to provide reasonable assurance that the CSSF would comply with the applicable dose limits in 10 CFR Part 20, include their documented radiation protection plan; safety analysis report; design, regulatory, and contractual enforcement mechanisms; as well as access controls, training and dosimetry.

Based on information DOE provided in its waste determination (DOE, 2023), NRC staff has reasonable assurance that the 10 CFR 61.43 performance objective for protection of individuals during operations can be met for the INL INTEC CSSF. NRC staff will monitor DOE's closure plans and activities as they progress to assess DOE compliance with the 10 CFR 61.43 performance objective.

4.4.2 Site Stability (10 CFR 61.44)

NRC staff reviewed DOE's waste determination (DOE, 2023) and noted that DOE indicated that to support demonstration of compliance with the 10 CFR 61.44 performance objective, that it planned to

- remove waste to the maximum extent practical
- stabilize remaining waste, equipment, and structures with grout

DOE also noted that

- the engineered barrier system provides structural stability and reduces contaminant migration to the environment

- the climate at INL INTEC CSSF is arid with low precipitation rates, and a large depth to groundwater

- the site is located in a remote location

- the site is geologically stable

NRC had technical comments that there was no specific discussion on a number of features, events and processes (FEPs) that could impact site stability in Chapter 7 of DOE (2023). A summary description of flooding analyses is provided in the waste determination, including a description of a flood analyses evaluated for the INL INTEC TFF PA (DOE-ID, 2003) involving an extreme precipitation event causing overtopping failure of the Mackay Dam and flooding of the BLR and INTEC. Chapter 7 also makes general statements regarding engineered features that provided long-term stability by limiting the amount of water infiltrating the waste bins and that provide a barrier to intrusion by burrowing animals, roots, or humans. High-level information is also provided regarding site and natural system features that promote site and geologic stability (e.g., low seismic activity). NRC requested a more thorough accounting of FEPs listed in NUREG-1854.

While DOE indicates that the engineered features will be filled with grout to limit void space and promote site stability, no details are provided regarding the grouting strategy and final configuration of the disposal facility making it difficult for NRC staff to assess DOE disposal facility compliance with the 10 CFR 61.44 performance objective. More detailed design information related to the bin sets and ancillary equipment, including information regarding potential void space that cannot be easily grouted; the intended grout formulations and associated performance requirements; and final disposal facility configurations, including final grade levels was requested.

In RAI SS-1 (NRC, 2024), NRC staff requested that DOE provide additional details or cross-references to sections of DOE (2023) and DOE-ID (2022) that provide analyses or evaluate potential impacts of FEPs that may impact site stability. FEPs should include the following: erosion, seismicity, extreme weather events, differential settlement, plant and animal activity, natural resource exploitation, potential perched water occurrence in relation to the elevation of the bin sets, and climate change (or increased infiltration). While this information may be provided in the PA or supporting references, NRC staff should not have to infer the types of analyses that are used to support the demonstration of compliance with the site stability and other performance objectives; the basis for the compliance demonstration should be clearly provided in the waste determination.

DOE referred to NUREG-1854, Chapter 7, for factors important to the demonstration of compliance with the site stability performance objective. DOE provided a crosswalk between the NUREG-1854 FEPs to sections of the PA, DOE Orders, or otherwise individually addressed each item in Chapter 7. In most cases, DOE provided information, analyses, data, or programs to provide support for the site stability performance objective. In some cases, final closure plans are not yet available to address the FEP, but DOE provided information on plans to develop closure documentation in the future. Therefore, NRC recommends that DOE document key assumptions or commitments that are important to the compliance demonstration in the final waste determination.

Based on information DOE provided in its waste determination and RAI response (DOE-ID, 2025a), NRC staff has reasonable assurance that the 10 CFR 61.44 performance objective related to stability of the disposal site after closure can be met provided certain key assumptions or commitments are met (see Sections 4.5 and 4.6). NRC staff will monitor DOE's closure plans and activities as they progress to assess DOE compliance with the 10 CFR 61.44 performance objective.

4.5 NRC Review and Conclusions (Criterion Three)

NRC staff reviewed the DOE CSSF waste determination, PA, and supporting references against technical review considerations in NUREG-1854 and found that DOE disposal actions at the INL INTEC CSSF can meet NDAA WIR criteria provided the following assumptions are verified during monitoring:

- Grout will be effectively used to fill void space in the engineered system and designed to meet performance requirements consistent with PA assumptions.
- Grout will be well mixed with the waste. Alternatively, DOE can show that the assumption does not lead to a significant underestimate of the projected dose (e.g., solubility of key radionuclides in the waste is sufficiently low to meet the 10 CFR 61.41 performance objective).
- DOE's assumption that water flow through the SS bins is proportional to the fraction of the bin surface area breached (and that limited corrosion data can be used to calculate the fraction of the bin surface area breached over time) does not lead to a significant underestimate in projected dose. Alternatively, DOE can show that the assumption does

not significantly affect the compliance demonstration (e.g., other barriers can compensate for underperformance of the stainless-steel bins).

- The chloride content of water used to grout the CSSF will be low and other properties of the grout will not negatively impact engineered system performance (e.g., excessive shrinkage, high permeability, and low-quality grout).

- FEPs (e.g., backfill material properties, poor engineered cover design, enhanced infiltration, flooding, and presence of perched water) will not negatively impact engineered and natural system performance (e.g., lead to significantly higher infiltration rates into the engineered system or rapid transport rates through the vadose zone compared to what is currently assumed in the PA).

- The inventory is not significantly higher than what is assumed in the PA.

- Natural attenuation will be as effective as assumed in the PA for key radionuclides.

- Ancillary equipment (e.g., transport lines) within 3 m [10 ft] of ground surface will be removed as part of the closure process.

While DOE evaluated the impact of most of these assumptions on projected dose in the PA (DOE-ID, 2022) or follow-up RAI responses (DOE-ID, 2025a), due to the presence of multiple redundant barriers and complexity of the PA modeling, NRC staff will continue to evaluate resolution of key technical issues during monitoring as CSSF closure progresses.

4.6 Monitoring to Assess Compliance with 10 CFR Part 61, Subpart C

To ensure key assumptions are met, NRC staff will monitor DOE disposal actions to ensure the following among other activities (see Appendix A for additional details):

- NRC staff will monitor the effectiveness of waste retrieval operations to ensure the inventory is not significantly higher than assumed in the PA.

- NRC staff will monitor DOE’s efforts to provide support for the assumption that the waste is well mixed with the grout or will review additional information that shows why this assumption does not lead to a significant underestimation in the projected dose.

- NRC staff will monitor DOE’s efforts to obtain additional support for assumed long-term performance of the SS bins, including updates to its PA to address technical issues identified in this TER.

- NRC staff will monitor DOE’s selection of grout formulation and design specifications to ensure a high-quality grout designed to fill void space in the engineered system is developed. The grout formulation should also minimize preferential flow pathways through the system and perform consistent with PA assumptions. For example, a grout formulation or formulations should be designed that is flowable enough to fill void space, while balancing negative properties such as excessive bleed, shrinkage, high permeability and porosity.

- NRC staff will monitor closure and grouting operations to ensure grout is tested and placed in accordance with performance requirements, which are consistent with key assumptions made in the PA.

- NRC staff will monitor DOE's plans as they are developed with respect to the final configuration and grade of the CSSF with most of the bins currently located above-grade, including design and construction of an engineered cover to limit infiltration into the closed bin sets.

- NRC staff will review periodic monitoring reports and hydrologic studies to ensure assumptions regarding relatively low long-term infiltration rates and natural attenuation are valid.

- NRC staff will monitor DOE's closure plans as they progress to ensure that commitments are made consistent with PA assumptions, including removal of ancillary equipment within 3 m [10 ft] of ground surface.

5.0 OVERALL CONCLUSIONS

As discussed in detail in previous sections of this TER, NRC staff has conducted a technical analysis of DOE-ID's waste determination for the CSSF at the INL. While DOE thinks that the stabilized residual calcine in CSSF bins and associated ancillary equipment are Class C or less at the time of CSSF closure, to take full advantage of the consultation process under the NDAA, DOE requested consultation with NRC regarding its disposal plans under Section 3116 (a)(3)(B)(iii). The NRC staff concludes that DOE-ID has adequately demonstrated that NDAA criteria in Section 3116 (a)(1), (a)(2), and (a)(3)(B)(iii)[30] can be met for residual waste disposed of at the CSSF based on DOE's responses to NRC's RAIs (DOE-ID, 2025a); supporting references; and information provided during meetings between NRC and DOE if certain key assumptions are verified during monitoring (see Section 4.5). The NDAA requires NRC, in coordination with the State of Idaho, to monitor disposal actions taken by DOE to assess compliance with the performance objectives in 10 CFR 61, Subpart C. NRC will continue to coordinate with the Idaho Department of Environmental Quality to develop a program by which NRC and the state will monitor DOE's disposal actions.

It should be noted that NRC staff is providing consultation to DOE as required by the NDAA, and the NRC staff is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW. This NRC staff assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental waste determinations at INL or other sites.

[30] Section 3116 (a)(3)(B)(iii) is only required by the NDAA for Greater than Class C waste.

6.0 RECOMMENDATIONS

Inventory

NRC recommends that DOE consider sampling of the residual waste after waste retrieval to provide a better estimate of the remaining CSSF inventory following closure. Sampling would help verify and validate the assumptions[31] made to develop the inventory. Composite sampling could be used to ensure the most representative sample without the need to analyze individual increments, making up the composite samples.

Corrosion Support

An important aspect of the DOE analysis is the lifetime of the SS bins. The long lifetime is dependent on the following:

- Formation of passive oxides on the steel surface and the preservation of passivity in the long-term,
- High pH solutions with low chloride concentrations in contact with the steel, while the grout is protective,
- Low chloride concentration environments after grout is degraded.

NRC staff recommends designing a monitoring system, in support of a closure strategy, to track the chemistry of infiltrated waters and environments in contact with the grout and steel, and to monitor degradation of SS bin materials. Placing probes in contact with the steel bins may not be desirable; probes and grout openings may have uncertain long-term effects. A safer alternative may be designing coupons of SS to mirror conditions of actual bins, surrounded by grout and simulated calcined solids. Electrochemical probes may be attached to the coupons to track parameters in long-term tests such as the corrosion electric potential and corrosion currents to verify passivity assumptions. The chemical composition of moisture and any potential deposits around coupons may be tracked to measure pH and chloride concentrations. The grout-steel coupons may be placed in soil in orientations designed to maximize contact with infiltrating water. The grout may be designed with cracks and gaps to explore effects of grout defects on infiltrating water and potential development of deposits in cracks and grout gaps. Moisture levels in surrounding soils may be tracked to correlate the extent of bin material corrosion attack to the moisture level in soil. These efforts would help strengthen the assumptions regarding the long lifetime of SS bins.

Radionuclide Transport

NRC staff recommends an updated assessment of available information on partition coefficients for radionuclide transport modeling that takes advantage of multiple lines of evidence, including laboratory and field studies and monitoring well data.

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[31] Uncertainties include (i) limited data on liquid waste samples prior to calcination being used to estimate the inventory, (ii) use of ORIGEN modeling and scaling factors to estimate the inventory for those radionuclides not directly measured in liquid waste, (iii) residual volumes based on expected waste retrieval operations, and (iv) assumption that residual waste is a homogenous mixture of the various waste streams that went into the bin sets, among other potential factors influencing uncertainty in the waste inventory for key radionuclides.

Biosphere Parameters

NRC staff recommend that biosphere parameters be updated to reflect more modern sources of information to support future PAs at the site, particularly for site-specific and potentially risk-significant parameters such as distribution coefficients and transfer factors.

Performance Assessment Update

In future PA updates, NRC staff recommends that DOE more fully address technical issues in this TER including the impact of assumptions that (i) the waste is well mixed with the grout, and (ii) the percent area of the SS bin breached based on limited corrosion data is proportional to flow through the waste zone.

7.0 CONTRIBUTORS

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APPENDIX A—MONITORING

If DOE finds that the waste associated with CSSF bin sets and ancillary equipment is found to be non-high-level waste, NRC will monitor DOE disposal actions to assess compliance with the performance objectives found in 10 CFR Part 61, Subpart C.

NRC expects DOE to keep NRC apprised of its closure activities to allow NRC staff to plan monitoring responsibilities to fulfill its responsibilities under the NDAA. The following documents are requested when they become available:

Periodic Documents

- Environmental Monitoring and Hydrologic Study Documents Pertinent to INTEC
- RCRA Closure Documents Pertinent to CSSF Closure

Final Design Documents

- Engineered Closure Cap Design
- Grout Formulation(s)
- Final End State Configuration

Table A-1 Provides a List of Monitoring Areas and Factors related to the 10 CFR 61.41 performance objective.

Monitoring Areas and Factors Related to the 10 CFR 61.41 Performance Objective

Monitoring Area	Monitoring Factor	Close-out
MA 1 Inventory	NRC staff will monitor DOE's waste retrieval efforts and development of final inventory estimates (i.e., concentration and volume) to ensure that assumptions regarding the residual inventory in the bin sets and consequently, potential dose, is not underestimated.	NRC staff will close this monitoring factor after the final bin set inventory is developed.
MA 2 Engineered Barrier Performance	Steel bin set performance—the performance of the SS bin sets is a key barrier in DOE's performance assessment. NRC will evaluate DOE methods to provide support for the long-term performance of the SS bins, including PA assumptions regarding the impact of calculations of percent bin area breached on flow through the waste zone.	NRC staff will close this monitoring factor after support for SS bin performance is obtained or the risk associated with the final inventory is confirmed to be acceptable under more aggressive corrosion conditions.
	Grouting of steel bin sets and vaults—the performance of the grout is important to ensuring that preferential pathways through the system that could enhance corrosion and lead to higher release rates from the disposal system will not develop. NRC staff will also evaluate DOE's grouting of the bin sets to ensure a passive service environment is created (e.g., low chloride content water is used). NRC staff will also monitor the final grout formulation to ensure that the chemical performance of the grout is not overstated in the PA.	NRC staff will close this monitoring factor after it has reviewed DOE's closure grout formulation documentation, and the last bin set is operationally closed and grouted.
	Engineered cover system—NRC staff will evaluate DOE's design of an engineered closure cap to ensure that poor design does not lead to enhanced infiltration rates through the disposal system.	NRC staff will close this monitoring factor after the engineered closure cap design is evaluated and the construction of the closure cap is observed.
MA 3 Natural System Performance	Environmental Monitoring—NRC staff will review environmental monitoring reports that provide information on infiltration, potential for creation of perched water, and natural attenuation to ensure key assumptions regarding natural system performance are verified.	NRC staff will close this monitoring factor after it determines that assumptions regarding natural system performance are adequately supported.

MA 4 PA Maintenance	NRC will monitor DOE's efforts to address NRC staff technical issues in future updates to its PA, including the assumptions that the waste is well mixed with the grout, and support for the assumption that the percent area of the stainless steel bin sets breached, based on limited corrosion data, is proportional to flow of water through the waste zone.	NRC staff will close this monitoring factor after the technical issues identified in this TER with respect to the PA model are addressed through additional model support or updates to the PA.