UNITED STATES
NUCLEAR REGULATORY COMMISSION
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April 8, 2024

Kristen G. Ellis
Acting Associate Principal Deputy Assistant
Secretary for Regulatory and Policy Affairs
US Department of Energy
Office of Environmental Management
1000 Independence Ave., SW
Washington, DC 20585

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING DRAFT BASIS FOR SECTION 3116 DETERMINATION FOR CLOSURE OF THE CALCINED SOLID STORAGE FACILITY AT THE IDAHO NATIONAL LABORATORY SITE (DOCKET NUMBER PROJ0735)

Dear Kristen G. Ellis:

By letter dated October 20, 2023, the U.S. 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 to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System [ADAMS] under Package Accession No. Main Library [ML] ML23304A026). The DOE Submittal to the NRC was under Section 3116(b) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA) (i.e., NDAA Monitoring). By letter dated December 19, 2023 (ADAMS Accession No. ML23342A109), the NRC acknowledged receipt of the DOE Submittal.

NRC has enclosed a Request for Additional Information (RAI), which is a list of comments for which the NRC staff needs responses from the DOE before the NRC can complete its review. These RAI's are based on our risk-informed review of the Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility and the supporting documentation. As NRC continues its review of DOE documents and RAI responses, NRC may develop additional comments for which NRC may need a response from DOE.

The NRC requests that the DOE provide responses to the enclosed RAIs 60 calendar days of the date of this letter. The NRC is planning a public meeting with DOE to discuss the NRC RAIs and/or any preliminary DOE responses.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Agency Rules of Practice and Procedure," a copy of this letter will be available electronically for public inspection in the NRC public document room or from the publicly available records component of ADAMS. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

If you have any questions, then please contact Mr. Maurice Heath of my staff at 301-415-3137 or by email, Maurice.Heath@nrc.gov.

Sincerely,

Signed by White, Duane
on 04/08/24

Duane E. White, Branch Chief
Low-Level Waste and Projects Branch
Division of Decommissioning, Uranium Recovery
and Waste Programs
Office of Nuclear Material Safety
and Safeguards

Enclosure:
1. Request for Additional Information

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

Performance Assessment RAIs

RAI PA-1:

Comment:

Screening analysis assumptions do not appear to be adequately supported potentially leading to elimination of key radionuclides from detailed modeling. Screening analyses are typically pessimistic to ensure that potentially risk-significant radionuclides are not screened out from further detailed analysis.

Basis:

The PA indicates that a dose threshold of 0.04 mSv/yr (4 mrem/yr) is used to screen out radionuclides from detailed modeling with no basis provided for the screening threshold. As a benchmark, Consolidated Decommissioning Guidance found in NUREG-1757, Volume 2, Rev. 2, provides guidance on insignificant radionuclides (and pathways) relaxing the need for detailed analysis for the combination of radionuclides cumulatively contributing less than 10 percent of the dose standard (e.g., 10 percent or 25 µSv/yr [2.5 mrem/yr] of the 0.25 mSv/yr [25 mrem/yr] unrestricted release dose standard). Furthermore, the only pathway considered in the screening analysis is drinking water ingestion. No other groundwater dependent pathways are considered, nor are other pathways considered. While the drinking water ingestion pathway may dominate the risk, additional information should be provided to support this assumption.

Additionally, Table 2-23 indicates that the source thickness is 1.5 m (5 ft), while it has not been demonstrated that the source will be well mixed in the grout placed on top of the waste[1] and the thickness of the concrete base for CSSF 1 underneath the concrete base is only 0.6 m (2 ft). Furthermore, due to the age and service conditions of the concrete base slab underlying the bin sets, it is unclear that the performance of the base slab will be as assumed in the screening analysis. Retention of key radionuclides in the source zone is considered a key uncertainty in the PA. These uncertainties in engineered barrier performance should be addressed with conservative assumptions in the screening analysis.

The screening analysis not only considers attenuation in the cementitious materials comprising the engineered system, but also credits radiological decay and attenuation in the natural system leading to elimination of several radionuclides that may have been screened in using alternative approaches (e.g., see screening approach used in INL INTEC TFF PA [DOE Idaho, 2003]) where the concentration in the waste zone was compared to the screening level with no credit for transport time, decay and attenuation in the natural system). Additionally, the selected transport parameter values used in screening analysis do not always appear to be conservatively selected. PA RAI.4 below contains additional information regarding potential non-conservatisms in the screening analysis K_d parameter

[1] AEA Technology (2006) describes calcine as potentially cohesive and hardened and notes that during retrieval technology testing that there was some success breaking-up and clearing hardened calcine. AEA Technology (2006) also notes that it was possible to block the nozzle and some material proved to be “intractable”. NRC staff are unsure if hardened waste is 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.

values.

Finally, dispersion is not considered in the screening analysis, although for relatively short-lived radionuclides (short-lived relative to the travel time), dispersion could lead to earlier arrival and higher exposure concentrations. For example, if no or minimal credit is given to sorption and attenuation during transport, and dispersion is considered in transport analysis, then relatively short-lived radionuclide Sr-90 could be screened in.[2] Although engineered and natural system barriers are expected to attenuate releases of relatively short-lived radionuclides, the importance of these barriers to mitigating release of the highest activity and relatively mobile key radionuclides like Sr-90 should be further studied in detailed analysis. Sr-90 and Cs-137 may also be important to the intruder analysis as discussed in RAI 5. Finally, diffusion of key radionuclides and early release of short-lived radionuclides such as Sr-90 prior to significant decay could also lead to higher concentrations and dose and should also be evaluated in screening analyses if it could lead to radionuclides being screened in.

Path Forward:

Provide additional information to support the 0.04 mSv/yr (4 mrem/yr) screening threshold, and elimination of other groundwater and non-groundwater dependent pathways. Provide additional support for the credit given to engineered system performance in the screening analysis considering uncertainty in performance (i.e., assumption that the waste at the bottom of the bin sets is mixed in 1.5 m (5 ft) of grout or cement and the selected distribution coefficients for cementitious materials which may be aged and cracked). Consider comments in RAI PA.4 related to potential non-conservatisms in the screening analysis related to K_ds. Perform additional detailed modeling or sensitivity/uncertainty analysis on additional key radionuclides that could have potentially been screened in, as applicable. This information will also allow NRC staff to better understand key barriers to mitigate release of high activity, relatively mobile radionuclides from the CSSF.

[2] It is important to note that Sr-90 was screened in and was a key radionuclide for the INTEC TFF PA (DOE Idaho, 2003) and that the inventory in the INL CSSF (1.17E+03 TBq or 31,700 Ci) is substantially greater than the total inventory in the INTEC TFF (0.866 TBq or 23.4 Ci maximum for a single tank [with 7 large tanks] and 9.2 TBq or 249 Ci in a sand pad [2 potentially contaminated sand pads WM-186 and WM-187]). Sr-90 was screening in because the screening analysis for the INTEC TFF was in one key respect more conservative compared to the screening analysis for the INTEC CSSF (e.g., the screening analysis for INTEC TFF did not take into consideration radiological decay during transport to the point of exposure). Likewise, Sr-90 was screened in for the F-Area Tank Farm Facility PA (SRS-REG-2007-00002, Rev. 1) due to conservatism in the screening approach.

RAI PA2:

Comment:

The assumed corrosion rates for stainless steel (e.g., Table 4-5 of DOE/ID-12008) were incompletely justified. The corrosion rates were measured in soil under very benign conditions that promote passivity. Many stainless steel test 304L coupons did not exhibit corrosion after exposure to soil for 12 years. Non-zero corrosion rates were only detected in a few coupons, summarized in Table 4-5. The performance assessment, using a distribution of corrosion rates based on statistics of the very few non-zero measurements in Table 4-5, optimistically assumes that stainless steel remains passive for hundreds of thousands of years. The analyses should account for other possibilities consistent with information in the stainless steel corrosion literature, such as corrosion rates in grout and stainless steel exhibiting localized corrosion such as crevice corrosion.

Basis:

Many of the stainless steel (304L and 316L) coupons exposed to soil in the 12-year testing at the Idaho National Laboratory Site exhibited no corrosion.[3] Table 4-5 of DOE/ID-12008 summarized the SS 304L tests with non-zero corrosion rates (i.e., on the order of a few nanometers/yr) that are consistent with passive corrosion. The soil exposure testing was not designed to represent a "bounding" set of conditions (e.g., more aggressive conditions than occur in pristine or degraded grout).

Passive corrosion rates for 304L SS are commonly reported on the order of 0.1 microns/yr in the literature, based on short-term electrochemical tests (e.g., https://doi.org/10.1016/j.jalmes.2023.100028, https://doi.org/10.1016/B978-0-08-103003-5.00008-4).

Corrosion rates for 304 SS in concrete porewaters have been measured and range from fractions of microns per year (e.g., http://hdl.handle.net/1808/20707), to 3 microns/year in a 2-yr exposure period experiment (https://doi.org/10.1016/j.conbuildmat.2019.06.120).

Stainless steel 304L is susceptible to crevice corrosion (e.g., doi:10.3390/ma15093055), when enough salt is present in the environment. Crevices could develop after grout degrades. Complex corrosion patterns could arise, with anodic and crevice zones exhibiting higher localized corrosion rates.

Coupons of stainless steel 304 exhibited signs of surface attack, pitting, blistering, and a type of corrosion referred to as "tunneling" corrosion after 33 years of exposure, especially on sensitized metal coupons (DOE/ID-11421 Revision 2, February 2018). Because of these data, prior iterations of the performance assessment credited only few decades of performance to the bins.

Path forward:

A more complete assessment is needed for credible degradation processes of stainless steel instead of postulating that SS passive conditions will prevail for hundreds of thousands of years. If stainless steel degradation processes are deemed unlikely or not possible, corresponding reasons should be clearly documented. If bounding corrosion rates are

[3] Adler-Flitton, M. K., M. E. Delwiche, and T. S. Yoder, 2011, *Long-Term Corrosion Degradation Test – Twelve-Year Results for the Idaho National Laboratory Site*, RPT-750, Rev. 0, Idaho Cleanup Project, August 2011.

adopted, those corrosion rates should be justified in the context of relevant studies available in the literature including analyses in DOE/ID-11421.

<u>RAI PA-3</u>:

<u>Comment</u>:

The scenario analysis related to degradation of the grout and steel bins appears very limited. In the absence of empirical data, uncertainties in corrosion, grout degradation processes, and water pathways appear high. Consideration of alternative and reasonable bounding scenarios is warranted to address those uncertainties.

<u>Basis</u>:

The conceptual model for the CSSF vaults assumes that the groundwater release pathway starts with advective transport of dissolved radionuclides from the grout into the concrete base, then the base provides additional retardation before release to the underlying basalt layer in the model. Advective release is limited by the following primary barriers: (i) complete protection from flow through the calcine waste layer until corrosion degrades the entire steel bin thickness, (ii) partial reduction in the fraction of grout exposed to flow while the area of complete steel degradation increases, (iii) reduction in flow through the calcine layer while the monolithic grout degrades[4], and (iv) low background infiltration flow after an initial period of 50 years before vegetation is established. The release model assumes that the grout is monolithic (no preferential pathways exist, such as grout shrinkage gaps or partings between grout layers), flow is vertical through the vault, and the cumulative fraction of waste contacted by flow is proportional to the cumulative fraction of the steel that has been corroded through (i.e., ongoing corrosion provides a series of continuous flow paths linking distinct entry and exit points instead of flow spreading throughout the grout).
The combined effect of the assumed barriers to release is to spread the model releases over hundreds of thousands of years in the base case. The steel corrosion model assumes that the 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. With these time frames, the model assumes grout degrades to sand before the steel begins to fail.

The sudden failure scenario of "one factor at a time" (OFAT) scenarios 1 and 2 suggest that establishment of rapid and widespread release may be the most significant challenge to performance. In such scenarios, geometrical details become more important; for example, the difference in travel times through a 0.608 m (2 ft) concrete base for CSSF 1 and the assumed 1.5 m (5 ft) concrete base assumed in the analysis may be insignificant when releases are spread out in time with steel degradation assumed to occur over hundreds of thousands of years; however, the assumed performance of the grout and underlying concrete base may be more significant for rapid events occurring over decades to centuries.

A brief qualitative analysis (i.e., not based on site-specific or condition-specific empirical data) was provided in paragraphs 3.2.2.3.1 to 3.2.2.3.7 of DOE/ID-12008 to exclude corrosion degradation processes for the steel bins. It appears feasible that grout may supply enough excess water to the calcine waste layer to promote inside-out corrosion at the bottom of bins, where contamination resides. Also, crevice and pitting corrosion could arise wherever crevices form, and especially at the bottom of the bins in contact with the vault base. Scenarios appear feasible that have preferential degradation at the bottom of bins with

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

much earlier corrosion penetration than currently computed[5], based on information in the literature on grout corrosion, crevice and pitting corrosion, and calcine corrosion of stainless steel. Absent condition-specific empirical data on steel degradation, it appears reasonable to assume 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.

The conceptual model for the vaults is that flow is vertical through monolithic grout and concrete base. The description of limited ingress from the top of the vault suggests that much of the water at the base of the vault would enter from the sides, implying a radially inward flow component within the vault and greater exposure of the bins to moisture on their exterior side. Shrinkage gaps and partings between grout layers may allow relatively fast penetration of water to the base of the bins during seasonal infiltration events (e.g., during snowmelt events).

DOE/ID-12008 does not describe the geometry of the bin bottoms. It is 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. Images in DOE/ID-12008 and EDMS ID 7188224 suggest that the bin bottoms are rounded and construction features exist that may create a grout-free gap between the bin bottom and the concrete base.

Two potential pulse release scenarios may be envisioned in conjunction with elevated corrosion rates near the base of the bins: (i) accumulated ingress of water from a hole in the sidewall that pools inside the bin until a second hole develops on the base or low on a sidewall, allowing the accumulated pooled water to drain within a few decades; and (ii) sufficiently widespread degradation of the bin base that exposes the base to episodic flooding events.

<u>Path forward</u>:

A more complete assessment of scenarios with faster degradation of stainless steel under complex environments is needed to support the compliance demonstration, especially if those scenarios cannot be ruled out based on information in the literature or condition-specific experiments. Also, attention should be given to scenarios with pulse releases[6]; for example, a scenario with a penetration that allows water accumulation inside the bin, followed by “flushing” after a penetration at the bottom of the bin occurs.

[5] Current performance assessment computations in DOE/ID-12008 postulate corrosion rates on the order of nanometers/year based on limited soil exposure experiments, which are considered unrepresentative of complex conditions under degraded grout and steel in contact with calcine waste.
[6] Pulse release scenarios may be more risk-significant compared to slow and prolonged releases assumed in the base case.

RAI PA-4

Comment:

The level of performance of the engineered and natural system is not fully supported. The cementitious material and natural system distribution coefficients or K_ds may not be conservatively selected leading to issues with the (i) screening analysis (i.e., screening out radionuclides that could have been screened in for more detailed modeling and sensitivity analysis), (ii) the base case compliance demonstration, and (iii) the selection of K_ds for the uncertainty analysis to study the impact of uncertainty on the results.

Basis:

It is not apparent that a critical analysis was conducted of K_d values adopted from prior compilations. The curium K_d of 4,000 L/kg used in the Phase II groundwater pathway screening analysis appears to be non-conservative relative to the 400 L/kg recommended in Prikryl and Pickett (2007). The latter recommendation was considered conservative, as it represented the lower end of the range of 400 to 1,000 L/kg recommended by Rodriguez et al. (1997) for curium by analogy with site-specific americium data. The PA reference for the 4,000 L/kg value was Jenkins (2001), which is not available to the NRC reviewers and was the source for nearly all K_d values used in the Phase II groundwater pathway screening analysis (PA Table 2-24).[7] The curium K_d values in PA Table 2-24 for both source and vadose zone are much higher than corresponding values for other actinides.

Some natural system K_ds utilized in the analysis were significantly higher than reasonably conservative values recommended in Prikryl and Pickett (2007) (see Table 1 for examples). Furthermore, there was no apparent consideration of the potential transport impact in the vadose zone of leachate derived from the waste facility (see discussion in DOE, 2011).

Table 1 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

For K_d values for cementitious materials, the PA references an evaluation performed for the Hanford site (WRPS, 2016; RPP-ENV-58782 Rev.00) that was based primarily on sorption databases from the Swiss and Swedish radioactive waste programs (NAGRA NTB 02-20 and SKB R-05-75, respectively). The WRPS evaluation appropriately chose values for oxidizing conditions and selected lower values when literature sources differed significantly. The PA evaluation, however, used values only for cementitious material degradation Stage II, stating that their uncertainty ranges will incorporate Stage I values. A basis was not provided for excluding consideration of K_d values appropriate for Stage III. Using a single set of values for cementitious materials neglects the potential effects on K_ds of chemical evolution (e.g., pH) during the different stages of degradation of the cementitious materials. The grout Np K_d values in PA Table 4-11 do not match those in the cited Table 6-5 in WRPS

[7] NRC is requesting this reference and may have additional comments after review of the reference.

(2016; RPP-ENV-58782, Rev. 0).

For uncertainty analyses of K_d values, it is unclear that the adopted statistical ranges adequately bound the uncertainties typical of geochemical processes (PA Table 6-3). As noted above; however, the NRC staff is concerned that adopted mean values for interbed sediments may be overly optimistic with respect to radionuclide transport. Any widening of uncertainty bounds using the adopted mean values will lead to also sampling optimistically high K_d values. Therefore, the technical bases for the adopted mean values need strengthening to provide confidence that potential dose will not be underestimated. Furthermore, the basis for using a GSD of 2 for Np and Se was not apparent and it is unclear that the GSD of 1.83 for U adequately captures uncertainty. Moreover, it is unclear why the K_d uncertainty analysis was limited to Np, Se, Tc, U, and Th and did not include potentially risk-significant radioelements such as I and Pu.

Regarding the grout K_d distributions for the uncertainty analysis (DOE/ID-12008, Table 6-3), the listed distributions are not consistent with the assumption that minimum and maximum values in Table 4-11 of the PA correspond to 0.01 and 0.99 quantiles of the log-normal distribution. The GSD values are not consistent with Eq. 6-1 in the PA; however, it is also noted that Eq. 6-1 appears incorrect. The GSD computed with Eq. 6-1 is not consistent with a log-normal distribution fitting the 0.01 and 0.99 quantiles at the provided values of the minimum and maximum K_d. Instead, the GSD can be simply computed by assuming that the minimum K_d corresponds to the 0.01 quantile of the log-normal distribution:

$$\ln GSD = \frac{\sqrt{2}}{3.29} \ln\left(\frac{GM}{\min Kd}\right)$$

Finally, the cementitious materials sorption coefficients used to attenuate releases from the waste zone and concrete base slab underlying the waste zone may overestimate the performance of the engineered system. The waste may not be fully mixed in the bottom 0.304 m (1 ft) of grout as assumed in the PA. Additionally, CSSFs are represented with a 1.5 m (5 ft) concrete base slab but CSSF 1, which generates the largest peak doses, has a 0.6 m (2 ft) concrete base slab. Adsorption in the base slab may be a substantial barrier slowing releases of key radionuclides.

Path Forward:

Clarify whether the correct values from WRPS (2016) were used in the PA. The WRPS best estimate value (100 L/kg) was used in the uncertainty analysis according to Table 6-3 in the PA, DOE/ID-12008 (2022).

DOE Idaho should address the following information requests concerning the use of distribution coefficients in the PA.

- Justify the high curium K_d values used in the Phase II groundwater pathway screening analysis (PA Table 2-24).

- Provide Jenkins, T., U.S. Department of Energy Idaho Operations Office, letter to Marty Doornbos, Idaho National Engineering and Environmental Laboratory, July 3, 2001, "K_d Values for INTEC Groundwater Modeling," EM-ER-01-115.

- Provide the basis for not considering K_d values appropriate for cementitious materials degradation Stage III; and lack of consideration of changes in K_ds over time due to chemical evolution (e.g., pH) during the different stages of cementitious material degradation.

- Evaluate the level of confidence that the selection of K_d values for sediments will ensure that groundwater pathway models will be unlikely to underestimate potential dose. Address whether a critical analysis of selected values was conducted to ensure overly optimistic values were not used.

- Provide the basis for neglecting the potential for cement leachate impacts on radionuclide mobility below the waste facility.

- Provide the basis for the interbed K_d distribution functions adopted for the uncertainty analysis (PA Table 6-3).

- Provide a basis for the grout K_d distribution functions in Table 6-3, including consistency of those distributions with the minimum and maximum values in WRPS (2016), Table 6-5.

- Consider an uncertainty analysis of the grout and interbed K_d values for I, given that the potentially complex geochemical behavior of I in the cementitious materials and subsurface could lead to relatively unretarded transport.

- Consider an uncertainty analysis of the grout and interbed K_d values for Pu, given its geochemical complexity and the potential for a mobile fraction and colloidal effects.

- Assess the impact of the assumption that the waste is mixed in the bottom 0.3m (1 ft) thickness of grout with respect to concentrations and dose at the receptor well. Assess the impact of assuming the CSSF 1 base slab is 1.5 m versus 0.6 m.

RAI PA-5

Comment:

The intruder analysis requires additional technical justification related to the biosphere parameters and assumptions. Additionally, insufficient detail is provided on results of the analysis to allow NRC staff to review the results of the analysis and identify key risk drivers.

Basis:

A key assumption in the intruder assessment is that all ancillary equipment within 3.04 m (10 ft) of the ground surface, will be removed. This assumption eliminates consideration of an excavation scenario in the intruder analysis, which if considered could have led to substantially higher potential intruder doses due to the larger quantity of radioactivity brought to the surface. 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 public could be exposed. Additional information regarding planned closure activities (e.g., binding commitments to remove all impacted infrastructure within 3 m or more of the ground surface), controls, and barriers to intrusion to prevent excavation into the transport lines following closure of the bin sets to support elimination of a more risk-significant excavation scenario from consideration in the PA would be beneficial to NRC's evaluation of the 10 CFR 61.42 performance objective. Also, additional information regarding the potential historical accumulation or plugging of transfer lines would allow NRC staff to better assess potential risk from intrusion into the transport lines.

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. Sensitivity analysis on the timing of intrusion into the transport lines could be provided to provide additional confidence that the 10 CFR 61.42 performance objective could be met. While the potential doses shortly after the 100 year institutional control period are expected to be extremely high, information already provided by DOE on the low likelihood of exposure to transport lines (see Chapter 7 of DOE/ID-12008) can be used to evaluate the overall risk of exposure to high activity CSSF waste after the end of the institutional control period.

NRC review of a subset of the biosphere parameters supporting the intruder assessment reveal a potential for underestimate of dose considering site-specific information or more modern sources available in the literature for various exposure pathways. For example, Case et al. (2000) and Rupp (1980)[8] are cited for several biosphere parameters. Review of Case et al. indicates that dated references (e.g., U.S. NRC (1977)[9] were used to support the selection of several of the biosphere parameters. Additionally, Table 7-5 of the CSSF PA indicates that several RESRAD-ONSITE default parameter values were used in the analysis. As stated in NUREG-1757, Volume 2, Rev. 2 (U.S. NRC, 2022), RESRAD-ONSITE default parameter values are not acceptable for use without further technical support. More recent guidance found in US EPA Exposure Factors Handbook (2011) and other sources could be used to support more site-specific parameters. Alternatively, mean behavioral, and metabolic parameters found in NUREG/CR-5512, Volume 3, could also be used with minimal

[8] Age-Dependent Values of Dietary Intake for Assessing Human Exposures to Environmental Pollutants.

[9] Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Rev. 1.

justification in the lieu of site-specific values.

The intruder analysis consider distribution of drill cuttings over a 2200 m^2 area, leading to a thickness of contamination of 1.3 cm (0.5 in) for the acute exposure scenario and over a 2200 m^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. The assumed area may lead to an underestimate of dose for certain radionuclides and pathways (e.g., Cs-137 dose could be higher with a smaller area and larger thickness of contamination). Distribution of drill cuttings over such a minimal thickness is not considered realistic and may lead to an underestimate of dose. Sensitivity analysis on the area and thickness of contamination should be performed to ensure the doses are not underestimated.

Detailed discussion of results of the intruder analysis are not provided in the PA. A more detailed discussion would put NRC in a better position to understand risk associated with inadvertent intrusion into the CSSF. Furthermore, no sensitivity analysis was provided on key parameter and assumptions[10]. While uncertainty in parameters could have been managed with conservative assumptions, certain assumptions and parameters are not demonstrably conservative as indicated above.

Path Forward:

Provide additional information (or point to information) regarding the potential for accumulation of waste in the transport lines (i.e., are the accumulations depicted in Figure 2-42 in the PA (DOE/ID-12008) and Figure 2-44 in the WD (DOE-ID-2022-01) based on real data or simply based on process knowledge of the most likely areas for accumulation (see figure below). [11] Indicate if there will be any effort to characterize transport lines in the future to estimate potential residual volumes remaining in the transport lines as part of the closure process. Indicate if inventory estimates are available for the transport lines other than those calculated based on an assumed volume of waste from the associated bin that might remain in the transport lines, which were calculated for use in the intruder calculations.[12]

Provide additional information regarding planned closure activities to remove infrastructure within 3 m (10 ft) or more of the ground surface, potential plans to construct intruder barriers, and any controls to deter potential human intrusion into the disposal facility and support elimination of a basement excavation or large construction project scenario. Perform additional analyses considering the comments in this request for additional information and provide additional discussion of intruder analysis results[13] by key radionuclide to provide additional support for the intruder analyses (e.g., timing of intrusion, selection and uncertainty in biosphere parameters, waste distribution assumptions following intrusion). Alternatively, DOE Idaho could provide additional support for deterministic parameters selected for the analysis.

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[10] Sensitivity analysis on the volume of waste in the transport lines was considered in EDF-11455.

[11] EDF-11119 (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.

[12] Archibald and Demmer (1995) estimates remaining volumes for the waste calcine facility based on an assumed coating of waste of 0.35 mm 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.

[13] Tables of results are provided in EDF-11132, Rev. 5, and EDF-11455, Rev. 1, but additional discussion and analysis of the raw output should be included.

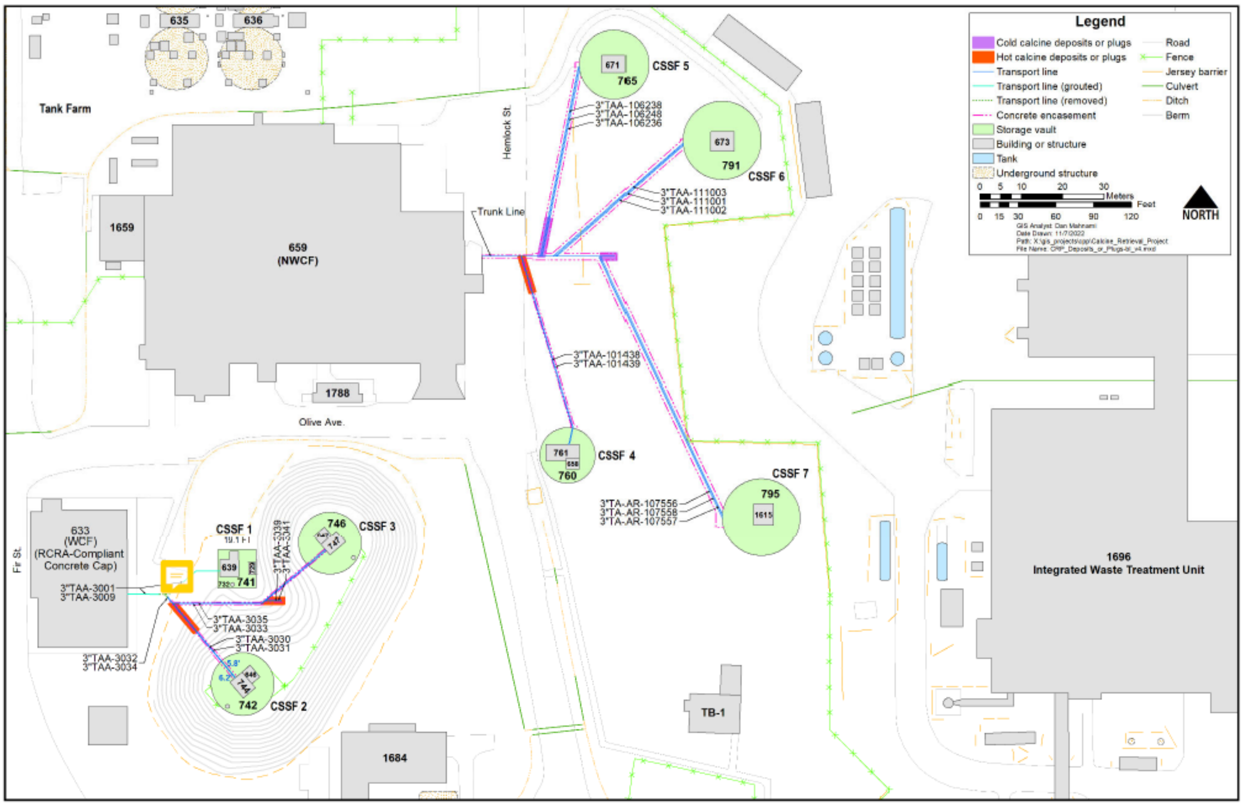

Image Credit: Figure 2-44 in DOE-ID-2022-01.

RAI PA-6

Comment:

The uncertainty analysis does not appear to capture the range of uncertainty in the results based on the uncertainty in the parameter values, and potential alternative scenarios. An effort to identify and reduce uncertainty in key parameter values should be considered. While the OFAT sensitivity analysis provides useful insights, the OFAT also generally suffers from lack of consideration of multiple failures of the engineered system[14] which could provide useful information on the redundancy of the barriers and the importance of individual barriers to the results. Overly broad distributions, also appear to have led to very significant risk dilution in the results, which should be evaluated and mitigated.

Basis:

As indicated in RAIs 2 and 3, steel bin barrier performance appears to be overly optimistic in the PA. The treatment of steel bin failure in the PA was entirely based on postulating steel remaining passive for tens to hundreds of thousands of years, leading to delay in the releases of radioactivity and spreading out in time of those releases from the bin sets. For example, Table 6-3 in DOE/ID-12008 contains parameter distribution information for the bin lifetimes, which are based on limited corrosion rate measurements under very benign environmental conditions. Use of the CDF of failure times (assumed to represent degradation in time of a bin surface) to further decrease a nominally low infiltration rate through the waste zone serves to prolong releases over long time periods potentially leading to significant "dilution" of "risk" over time, which should be evaluated in the PA. Alternative scenarios of bin failure, including scenarios with early failures and significant widespread damage, should be evaluated to address issues of potential risk dilution.[15]

Additionally, 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; greater infiltration rates through the engineered system; and potential underperformance of a partially degraded concrete base slab in attenuating releases of radioactivity from the storage system).

Infiltration in the uncertainty analysis is only varied between 0.5 cm and 2 cm, although the base case value used in the INTEC TFF was 4 cm/yr considering the presence of an earthen cover with sensitivity analysis evaluating infiltration rates up to 12 cm/yr. Additionally, based on a review of historical groundwater data, relatively fast transport rates of radionuclides have been observed through the vadose zone suggesting (i) higher infiltration rates, (ii) episodic flow, or (iii) other mechanisms that could lead to faster transport rates, and higher concentrations and dose in the Snake River Plan Aquifer from releases from the CSSF. For example, capillary barriers[16] at interbed/basalt interfaces could lead to damp interbeds, ephemeral formation of perched water with resultant lateral flow, and focused flow in localized areas following higher than average infiltration events and stair stepping between interbed layers and rapid transport of radionuclides through fractures in the basalt. While faster transport rates from historical releases may have been associated with anthropogenic water sources or liquid waste releases, other mechanisms such as seasonal variations in

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[14] OFAT 1 does provide a failure scenario involving multiple barriers.

[15] See NUREG-1757, Volume 2, Rev. 2, Appendix Q for more about parameters with overly broad distributions affecting the time of peak dose leading to potential risk dilution.

[16] Capillary barriers could be a result of infilling of fractures at the top of a basaltic layer with fine materials or the entry pressure into relatively large fractures within the basalt layer.

flow including enhanced infiltration rates due to spring snow melt should be considered.

Uncertainty in the results due to parameter uncertainty does not appear to be rigorously evaluated. The uncertainty in the inventory is a factor of 2 for a limited number of radionuclides (Np-237, Se-79, and Tc-99) and may not consider several uncertainties. Some of these uncertainties include the following:

- The uncertainty in the volume and concentrations of waste remaining in the bin sets, particularly considering that retrieval has not been conducted for any bin set,
- Lack of validation of concentrations of key radionuclides in caline waste following calcination including hard-to-detects (e.g., only Cs-137 and Pu-238 results from liquid tank waste samples prior to calcination are provided in the PA; Staiger and Swenson (2021) indicates that few or no sample data exist for long-lived, low activity radionuclides such as technetium (Tc)-99 and iodine(I)-129)).
- Uncertainty in ORIGEN 2 modeling and resulting uncertainty in radionuclide vectors or scaling factors is not discussed or explicitly assessed[17],
- Uncertainty in the data on waste streams and variability of key radionuclide concentrations in the various waste streams, and uncertainty introduced with the assumption that the residual waste remaining in the bottom of the bin sets after retrieval is a homogeneous mix of all waste streams stored in the bins prior to waste retrieval,
- Uncertainty in the CSSF 6 inventory due to transfer of CSSF 1 waste to CSSF 6, and
- Data input and other potential errors not addressed in the PA.

Limited sampling is conducted to validate the waste inventory and are based on liquid samples taken prior to calcination and would not address issues regarding homogeneity of the waste remaining at the bottom of the tanks. Only 10 percent of the I-129 is expected to remain after calcination, and Np-237 and C-14 inventories were also adjusted, although a discussion regarding the uncertainty in the inventory estimates for these radionuclides is not considered (e.g., I-129 inventory is stated to be a bounding estimate but it is unclear how the uncertainty in inventories for Np-237 and C-14 is considered or managed).

Arguments are presented in EDF-11126 (2021) regarding difficulty in representative sampling and difficulty mixing the sample. However, the assumption that the residual waste is a well-mixed fraction of all of the waste that went into the stainless steel storage bins is unsupported given the expected method of waste retrieval, which pulls waste from the bottom of the bin creating a cone of depression at the point of waste retrieval suggesting that waste at the top of the bins will flow to the bottom and be removed from the bin before waste at the bottom of the bin. Therefore, a higher percentage of waste at the bottom of the bin may remain after waste retrieval operations are complete. Sampling of the residual waste after waste retrieval would provide a better estimate of the remaining inventory which would be more representative compared to the current approach. Reducing uncertainty in the inventory may be cost beneficial if variability in key radionuclide concentrations across waste streams is significant. Composite sampling could be used to ensure more representative sampling and a better estimate of the average inventory remaining in the bin sets without the need to analyze several discrete samples.

Engineered and natural system distribution coefficients (and central values of associated parameter distributions) may have been overestimated leading to an underestimate of dose. See RAI PA.4 for more information.

Uncertainty in biosphere parameters is not evaluated and the current parameter set appears

[17] Staiger and Swenson (2021) provides information about errors associated with ORIGEN-2 based radionuclide calculations used for trace species that cannot be detected in the waste and cites Wenzel (2005), which supports a factor of 2 uncertainty range based on ORIGEN-2 modeling.

to potentially underestimate potential doses to members of the public (see also discussion in RAI 5 on selected biosphere parameters for the intruder analysis and alternatives for mitigating uncertainty in behavior and metabolic biosphere parameters).

Path Forward:

Update the sensitivity and uncertainty analysis to provide more realistic estimates of potential uncertainty in multiple key parameter values and results considering the comments in this request for additional information. The updated analysis should consider additional radionuclides and parameters that may not have been initially studied.

Evaluate the realism and impact of the prolonged steel bin residual performance assumed in the base case and uncertainty analysis and the potential for multiple failures not currently considered (or more extreme failure such as chemical failure of the grouted waste form).

Evaluate higher infiltration rates than what is currently considered, as well as alternative conceptual models with focused flow and faster transport rates through the vadose zone.

Identify key parameters to dose and consider whether it is cost beneficial to reduce uncertainty in those key parameter values to reduce uncertainty in the compliance demonstration (e.g., representative and composite sampling of waste following waste retrieval to validate the inventory estimates). Provide additional information on why the calcine waste cannot be sampled following waste retrieval operations to reduce uncertainty in the inventory estimates. Reevaluate uncertainty in key radionuclide inventories considering a wider range of factors discussed in the request for additional information. Discuss whether hard to detect radionuclides were sampled in the waste which may be key dose drivers and the relative uncertainty in those radionuclides compared to other radionuclides.

Consider more realistic, site-specific mean sorption coefficients and parameter distributions as indicated in RAI PA.4.

Compare the peak of the mean to the mean of the peak to ensure that risk dilution is not an issue due to overly broad parameter distributions that affect the timing of peak dose such as K_d and stainless steel bin failure times. See NUREG-1757, Volume 2, Rev. 2, Appendix Q (U.S. NRC, 2022) for more information.

Provide groundwater pathway dose conversion factors for the PA to allow NRC staff to better focus its review of biosphere parameters and modeling for key radionuclides.

Comment Related to Site Stability (10 CFR 61.44)

RAI SS-1

Comment:

Insufficient information is provided in DOE-ID-2022-01 for NRC to evaluate the 10 CFR 61.44 performance objective, "Stability of the disposal site after closure."

Basis:

Chapter 7 of DOE-ID-2022-01 provides limited information regarding features, events and processes related to the disposal facility important to site stability. A summary description of flooding analyses is provided, 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 Big Lost River 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. General statements are also made regarding site and natural system features that promote site and geologic stability (e.g., low seismic activity). These general summaries or statements should be supported with more specific information.

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 performance objectives (PO) in 10 CFR Part 61, Subpart C. More detailed design information related to the bin sets and ancillary equipment, including information regarding potential void space that cannot be easily grouted would be supportive of NRC staff's evaluation. Information about the intended grout formulations and associated performance requirements, and final disposal facility configurations, including final grade levels would also be beneficial.

Details or cross-references to sections of DOE-ID-2022-01 and DOE/ID-12008, Rev. 0 that provide analyses or evaluate potential impacts of features, events and processes (FEPs) that may impact site stability should be provided. 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). Much of this information is provided in other sections of the waste determination or PA, or is provided in numerous references (e.g., EDF-11231, EDF-11201, and EDF-11173). However, 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 POs; the basis for the compliance demonstration should be clearly provided in the waste determination.

Although detailed closure plans are not available and closure alternatives and final configurations are still be considered, a general discussion on how site stability and long-term protection of members of the public, including inadvertent intruders, is factored into the decision-making process when deciding between various alternatives would also be beneficial.

Path Forward:

Refer to NUREG-1854, Chapter 7, for factors important to the demonstration of compliance with the site stability PO and ensure all of the technical review areas important to site stability are addressed in DOE-ID-2022-01 as indicated in this request for additional information (or provide a cross-walk to additional references which address the components of the site stability analysis).

Removal of Key Radionuclides to the Maximum Extent Practical RAIs

RAI MEP-1

Comment:

Insufficient detail is provided as to the specific iteration of the pneumatic bulk retrieval system will be employed. Insufficient detail is provided to justify the results of the full-scale mockup of one group of calcine storage bins within CSSF 1 to the bins within CSSF 2-6. The WD does not provide a quantitative description of systems considered for the residual cleanout system or criteria to trigger the development and implementation of such technologies.

Basis:

The WD states that a bulk retravel system using a pipe-in-pipe vacuum and compressed air system will be used to retrieve calcine from the bottom of the bins. A recent retrieval demonstration (Sandow 2021) is cited as a "full-scale integrated mockup bin is a replica of one group of calcine storage bins within CSSF 1, and the cyclone is a replica of the cyclone located at CSSF 6." The results of the demonstration indicate that the system may remove ~99% of calcine waste. However, the WD does not describe specifications and operating parameters of the retrieval system or the detailed results of testing. Likewise, the WD does not discuss, qualitatively or quantitative, the impacts of the full-scale mockup of one group of bins within CSSF 1 with the differences in bin configuration and geometry of bin sets 2-6.

The WD also cites multiple historical references of smaller-scale mock-ups from 1981, 1995, and 2005 as viable demonstrations that this technology can achieve removal of waste to the maximum extent practical. However, the WD does not discuss the differences in these historical mockups from removal in the actual calcine bins, including deviations in bin geometry; bin material composition; removal depth; total volume; additional stress on the pneumatic retrieval system due to pressure, volume, radiation effects; etc. Because these historical mockups provide some basis for the removal of 99% calcine, NRC staff are seeking to understand the validity of using these tests as a justification, given the many differences between the testing environment and the actual configuration of CSSF bins.

DOE INL may also use a residual cleanout system to remove further calcine, if necessary (Sections 5.1.1.3, 5.1.3, 5.2). However, the details of these additional technologies are not discussed in the WD. DOE INL provides a general description and cost comparison of developing two systems (robotic vacuum crawler and articulating arm) but later states "further removal of CSSF residual waste after the initial bulk retrieval… is not cost effective, and any efforts to remove more of the small quantity of waste remaining would increase risk to the workers and not significantly reduce the potential risk to the public and the environment" (Section 5.3). It is unclear if DOE INL has decided to consider the development of residual cleanout technology, or if specific criteria will be established to trigger the development of such technology in order to remove calcine waste to the maximum extent practical.

Path Forward:

Include additional information in the WD, regarding the technical specifications and capacity of the pneumatic retrieval technology that will be employed. Include specific results (volume of waste removed and volume and depth of waste remaining) of the full-scale mockup testing.

Provide 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.

Given the differences noted, clarify the use of results from testing the historical mockups (1981, 1995, and 2005) as a basis to support that the pneumatic retrieval system can feasibly remove calcine to the maximum extent practical.

Provide technical specifications, technical advantages and disadvantages, and testing results (if available) for the 2 proposed residual cleanout systems (robotic vacuum crawler and articulating arm). Provide discussion of whether these technologies will be employed, possibly including decision criteria to further develop and implement these technologies to remove calcine to the maximum extent practical.

RAI MEP-2

Comment:

Insufficient information is provided regarding the process to monitor performance of the pneumatic retrieval system during operation and criteria for terminating retrieval activities. The WD does not include details of how DOE INL plans to verify that retrieval operations have removed calcine and/or highly radioactive radionuclides (HRRs) to the maximum extent practical.

Basis:

WD Section 5.1.1 "Selection of Waste Retrieval Technology" notes that one of the objectives of testing the pneumatic system is to "establish criteria for ending calcine retrieval activities (i.e., determine when the maximum extent practical has been achieved)." However, the WD does not provide additional details regarding criteria established to determine when waste retrieval has proceeded to the maximum extent practical and retrieval operations can be terminated.

Testing from fall 2005 (AEA 2006) notes that 99% of calcine simulant was removed from a depth of 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 1 inch across the bottom of the tank. "Integrated Mockup Test Report – 2020 and 2021" (Sandow 2021) states that retrieval activities "with the manual air lance continued until minimal retrieval rates were occurring" during Bin A Bulk Retrieval Testing but doesn't specifically cite what those rates were that triggered cessation of operations. From these two tests, it appears that different criteria were used to assess when to terminate operations.

Although tests have demonstrated that 99% removal of calcine may be achievable with bulk retrieval with the pneumatic system, the objective of removal to maximum extent practical is to continue retrieval operations until there are diminishing returns related to worker and public dose and cost. The WD indicates that bulk retrieval of 99% of calcine, in aggregate, is consistent with the objectives of removal to maximum extent practical. However, NRC staff have pointed out technical issues and uncertainties in the dose estimates reported in the performance assessment which assumes 99% removal (or around 2 inches (5 cm) of residual calcine depth remaining in the bin sets). NRC staff are seeking specific criteria related to how DOE INL plans to evaluate performance of pneumatic system during operations. These may include periodically inspecting the system performance using the video monitoring or other capabilities to assess depth of residual calcine, change in calcine removal rates, etc.

Sandow (2021) notes that "the amount of material left over is highly dependent upon material properties including bulk density, particle size, and angle of repose." As such, it is risk significant to this review to more fully understand the method by which INL DOE will evaluate performance of the pneumatic retrieval system.

Additionally, NRC staff are seeking to understand how DOE INL plans to verify that retrieval operations have been successful in removing highly radioactive radionuclides (HRRs) to the maximum extent practical after operations.

Path Forward:

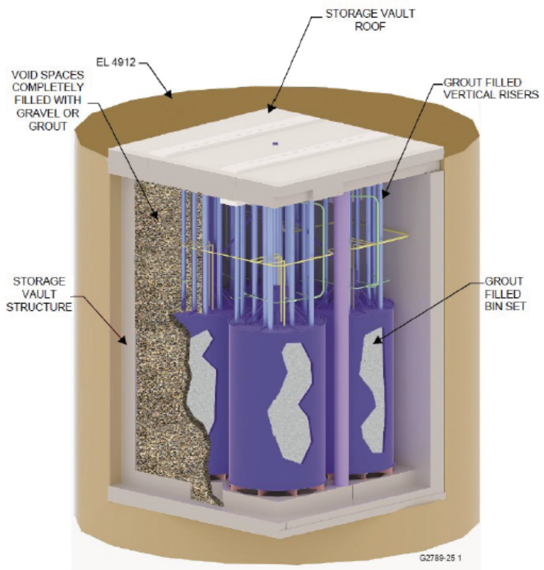
Provide the steps to the decision-making process to determine the termination of retrieval operations. This may include performance monitoring during retrieval operations and/or criteria for stopping retrieval activities.

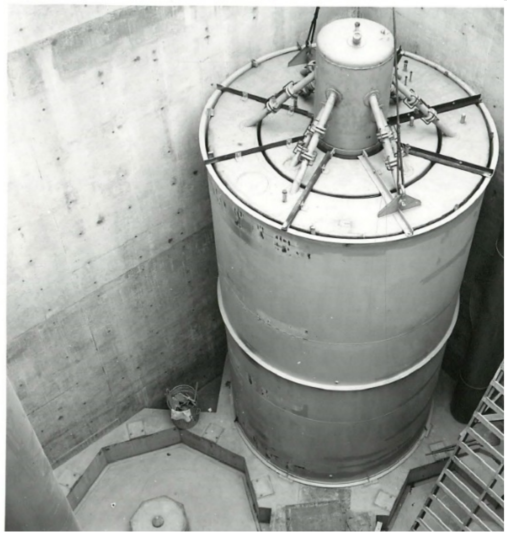
Provide the process of how DOE INL will verify that retrieval operations have successfully removed highly radioactive radionuclides (HRRs) to the maximum extent practical, as defined by the operations termination criteria developed by DOE.

Clarifying Questions or Comments

Note: The information requests above are rolled up to ensure that multiple issues that may affect the results of the compliance demonstration are addressed concurrently. The questions below are typically smaller questions for which NRC would like to obtain additional information. These questions are generally simpler to address compared to the RAIs. Some of the questions below may also be included in an RAI but are listed below to ensure the specific question is addressed.

1. Describe or point to a reference which provides information about the geometry of the base of the bins, including stands, pits/pedestals, sumps and other spaces that may be difficult to grout (or be important to near-field release model conceptualization). It is unclear if there are potential gaps/fast flow pathways or potential void volume after grouting that might facilitate radionuclide release.

<table>
<tr><th>Conceptual Model for Grouting of Bin Set (see space at bottom of bins)s</th><th>Potential Void Space Underneath Bins (see space on bottom left)</th></tr>
<tr><td>


</td><td>

</td></tr>
<tr><td>Image Credit: EDF-11231, Figure B-1. CSSF 1 conditions for Removal Action Alternative 1. EDF-11231, Calcine Disposition Project: Technical Feasibility Study for the Calcined Solids. September 2020.</td><td>Image Credit: EDMS ID 7188225 (page 29 of 337)</td></tr>
</table>

2. Provide additional information and references related to any events and pathways for water entry into the reinforced concrete vaults. For example, water leaked into CSSF-1 through underground piping or through joints between the roof and wall, or piping penetrations in the vault roof (see Section 2.11.1.1 in DOE-ID-2022-01). This is important to assess initial conditions and potential pathways for moisture into the bin sets.
3. Assess the potential for gaps, such as shrinkage cracks of grout away from tank walls that could provide potential flow pathways that bypass the stainless steel bins and grout (see Walter et al., 2009, Water et al., 2010, and U.S. NRC 2015) potentially allowing a significant amount of water to contact the steel bins enhancing corrosion, as well as provided a preferential release pathway through the waste zone and into the surrounding environment.

4. Describe the expected impact of extended heat generated by stored calcine on the hydraulic and geochemical properties of the base slab and vault walls. These barriers may have already substantially degraded, potentially allowing early enhanced flow towards the bins or less attenuation of releases of residual radioactivity from the waste zone through the base slab.
5. Assess the potential for localized rapid flow through the concrete base and potential consequences for transport (e.g., a gap between the bin and concrete base slab may allow flow to focus into cracks in the base, and the flow to then bypasses adsorption sites in the matrix). Transport rates through the base slab may be greatly enhanced by greater velocities and much less retardation.
6. Briefly clarify how the model for radionuclide leaching is implemented in the MCM model using equations 4-41 and 4-42. For example, Table 4-7 indicates that the grouted waste layer is 1.52 m, which may imply upwards diffusion of residual radioactivity into the overlying grout and may reduce release rates from the grout to the base slab.
7. Assess the potential for transient perching below the surficial alluvium and above the basalt layer from large snowmelt events, for example. Transient perching may provide a source of water that contacts the base of the steel bins, providing enhanced corrosion and release.
8. Provide the values used for effective diffusion coefficients in the grout, base slab, and natural system. Eq. 4-44 of DOE/IN-12008 (see page 250 of 534) implies that the effective diffusion coefficient under unsaturated conditions is orders of magnitude smaller than the diffusion coefficient in free water.
9. Clarify travel times and degree of pulse spreading of key radionuclides in various layers of the MCM model. Section 4.1.4.1 of DOE/ID-12008 discusses travel times around 278 years for non-sorbing constituents through the vadose zone; however, calculations using tabulated values in the PA suggest much longer durations.
10. Figure 5-1 in DOE/ID-12008 illustrates water flux through the grouted waste layer as a function of time for the CSSFs based on hydrologic failure of the CSSFs based on corrosion of the stainless steel bins and the thickness of the stainless steel bins. Figure 10 in EDF-11137 appears to show the same water flux through the CSSFs reflecting the hydrologic failure of the bin based on a distribution of corrosion rates and minimum thickness of the stainless steel bins. It appears that tunneling corrosion was considered as well as other tests with substantial stainless steel damage for EDF-11137 but then a decision was made to switch to more passive general corrosion rates. Explain the difference in the figures and the rationale for not considering tests with substantial steel damage, which resulted in shifting the failure times from hundreds to thousands of years; to millions to tens of millions of years.

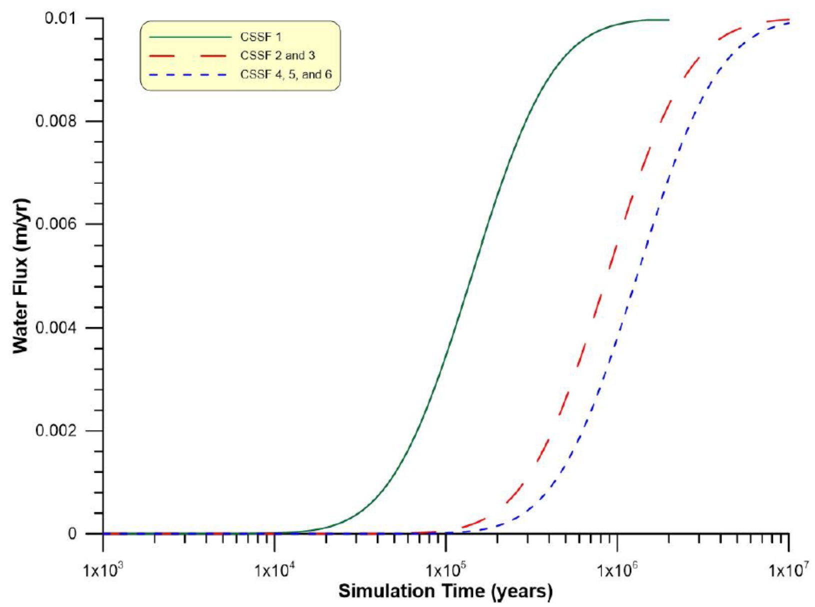


Image Credit: Figure 5-1. Water fluxes through the grouted waste layer as a function of time for the CSSF. These curves reflect the hydrologic failure of the bin sets based on a distribution of corrosion rates for Idaho National Laboratory Site soils and minimum thickness of the stainless steel bins.

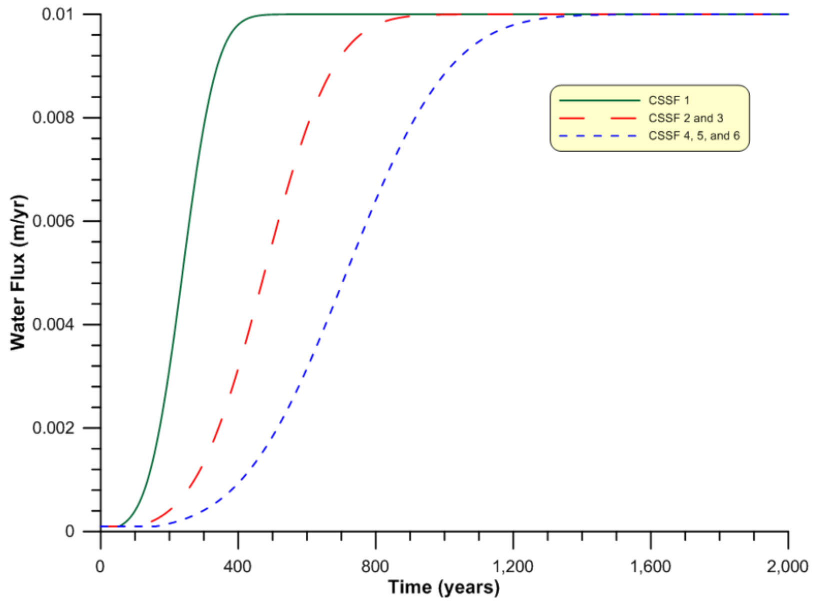


Image Credit: Figure 10 in EDF-11137, Rev. 1, Water fluxes as a function of time for the CSSFs. These curves reflect the hydrologic failure of the bin based on a distribution of tunnel corrosion rates and minimum thickness of the stainless steel bins.

11. Clarify whether the correct Np grout K_d values from WRPS (2016) were used in the PA, rather than the values in PA Table 4-11, DOE/ID-12008 (2022).
12. The Selenium K_d in grout parameter distribution appears to be in error (GSD of 195). Correct the error in the GSD for Se-79 if necessary (or clarify the typo).
13. Clarify if all remaining impacted infrastructure within 3 m (10 ft) of ground surface is being removed as part of the closure process. Clarify locations of transport piping that is not encased in concrete and whether transport piping will be grouted in place (e.g., transport piping from Structure 633 to CSSF 1 does not

appear to be encased in reinforced concrete according to Figure 2-43 in DOE/IN-12008 (2022) and Figure 2-44 in DOE-ID-2002-01 (2023) but does seem to be grouted in place).

14. Clarify the basis for the locations that indicate potentially plugged transport lines in Figure 2-43 and any efforts to characterize remaining transport lines and ancillary equipment.
15. Clarify other impacted ancillary equipment and components which will remain following closure and if any efforts will be made to develop an inventory for those components.
16. Provide discussion of how changes to the PA and/or intruder analysis (residual waste mixing in grout, volume of waste exhumed mixed with soil, etc.) as a result of these RAI responses impact the waste classification calculations in Section 6 of the WD.
17. Provide groundwater pathway dose conversion factors (mSv/yr per Bq/L) for radionuclides simulated in the PA (also add Sr-90) to facilitate NRC staff's review of the PA biosphere calculations.
18. Provide the modeling files for the (i) screening analysis, and (ii) deterministic base case to allow NRC staff to better evaluate the key assumptions and results of the PA modeling.

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