REQUEST FOR ADDITIONAL INFORMATION ON THE DRAFT SECTION 3116 DETERMINATION FOR IDAHO NUCLEAR AND ENGINEERING CENTER TANK FARM FACILITY (DOE/NE-ID-11226)

<u>Review Documents</u>

DOE-ID, 2003a, Idaho Hazardous Waste Management Act/Resource Conservation and Recovery Act Closure Plan for Idaho Nuclear Technology and Engineering Center Tanks WM-182 and WM-183, DOE/ID-10802, Rev. 1, September 2003.

DOE–ID, 2003b, “Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory,” DOE/ID–10966, Rev. 1, April 2003 (Errata December 2, 2003).

DOE-ID, 2003d, Composite Analysis for Tank Farm Closure, DOE/ID-10974, April 2003.

DOE–ID, 2005, “Draft Section 3116 Determination Idaho Nuclear Technology and Engineering Center Tank Farm Facility,” DOE/NE–ID–11226, Rev. 0 Draft, September 7, 2005.

ICP, 2004e, “Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM–185 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility,” ICP/EXT–04-00358, Rev. 0, July 2004.

ICP, 2005c, “Generation, Disposition, and Current Inventory of Radionuclides in the INTEC Tank Farm,” ICP/EXT–05–00928, June 2005.

INEEL, 2003, “Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM–184, WM–185, and WM–186 Tank Residuals,” INEEL/EXT–03–00057, Rev. 0, April 2003.

INEEL, 2004b, Data Quality Assessment Report for the Post-Decontamination Characterization of the Contents of Tank WM 183 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility, INEEL/EXT-03-01202, Rev. 1, July 2004.

Latchum, J. W., A. L. Biladeau, E. S. Brown, D. D. Deming, B. L. Schmalz, F. M. Warzel, and L. J. Weber, 1962, Report of the Investigating Committee, CPP Waste Tank WM-187 Leakage of March 17, 1962, La-53-62A, Phillips Petroleum Company, 1962.

PEI–EDF–1009, Rev. 2, Portage, Inc., Idaho Falls, Idaho, August 31, 2005.

Portage, 2005a, "WM-180 Tank Solid and Liquid Volume Estimates and Post-Decontamination Radionuclide Inventories," PEI-EDF-1019, Rev. 1, Portage, Inc., Idaho Falls, Idaho, August 31, 2005.

Portage, 2005b, "WM-181 Tank Solid and Liquid Volume Estimates and Post-Decontamination Radionuclide Inventories with Comparisons to Performance Assessment Modeling Results," PEI EDF-1016, Rev. 2, Portage, Inc., Idaho Falls, Idaho, August 31, 2005.

Portage, 2005c, “WM–182 Tank Solid and Liquid Volume Estimates and Post-Decontamination Radionuclide Inventories with Comparisons to Performance Assessment Modeling Results,” Portage, 2005d, "WM-183 Tank Solid and Liquid Volume Estimates and Post-Decontamination Radionuclide Inventories with Comparisons to Performance Assessment Modeling Results," PEI EDF-1010, Rev. 2, Portage, Inc., Idaho Falls, Idaho, August 31, 2005.

Portage, 2005e, "WM-184 Tank Solid and Liquid Volume Estimates and Post-Decontamination Radionuclide Inventories with Comparisons to Performance Assessment Modeling Results," PEI EDF-1011, Rev. 2, Portage, Inc., Idaho Falls, Idaho, August 31, 2005.

Portage, 2005f, "WM-185 Tank Solid and Liquid Volume Estimates and Post-Decontamination Radionuclide Inventories with Comparisons to Performance Assessment Modeling Results," PEI EDF-1012, Rev. 2, Portage, Inc., Idaho Falls, Idaho, August 31, 2005.

Portage, 2005g, "WM-186 Tank Solid and Liquid Volume Estimates and Post-Decontamination Radionuclide Inventories with Comparisons to Performance Assessment Modeling Results," PEI EDF-1013, Rev. 2, Portage, Inc., Idaho Falls, Idaho, August 31, 2005.

Portage, 2005k, "Sensitivity of Radionuclide Residual Waste Location in a Grouted Tank at the Tank Farm Facility," PEI-EDF-1022, Rev. 0, Portage, Inc., Idaho Falls, Idaho, August 2005.

Portage, 2005i, "Methodology for Evaluating Tank Sample Data and Evaluating Compliance with Performance Assessment Results," PEI–EDF–1021, Rev. 0, Portage, Inc., Idaho Falls, Idaho, August 2005.

Portage, 2005j, "Analysis of Sorption Coefficient (K_d) Values Supplemental Report for the Tank Farm Facility Preliminary Assessment," PEI–EDF–1023, Rev. 0, Portage, Inc., Idaho Falls, Idaho, August 2005.

Portage, 2005l, "Evaluation of ^{99}Tc Drinking Water Dose for Oxidizing Sorption Coefficient in the Tank Grout," PEI–EDF–1024, Rev. 0, Portage, Inc., Idaho Falls, Idaho, August 2005.

Portage, 2005m, "Radiological Dose Calculations for the Tank Farm Facility at the Idaho Nuclear Technology and Engineering Center," PEI–EDF–1020, Rev. 0, Portage, Inc., Idaho Falls, Idaho, May 24, 2005.

Wenzel, D. R., 2005, "Relative Inventories of Reactor-Produced Species in INTEC Waste Types," EDF CRPD-001, Rev. 2, February 24, 2005.

1 **Comment**:

As recommended in SECY–03–0079 (NRC, 2003), the U.S. Department of Energy–Idaho Operations Office (DOE–ID) should investigate methods for measuring or better estimating the contaminated sand pad radionuclide inventories.

Basis:

Table 7 of the Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM–184, WM–185, and WM–186 Tank Residuals (INEEL, 2003) provides anticipated sample collection points for the tanks as well as the vault sumps. The waste determination includes the following text:

> *"After cleaning, the water used to flush the vaults was pumped out using the steam jets in vault sumps. The sumps are generally 0.3 m (1 ft) square (some vaults have larger sumps). Review of the sampling data shows that the concentrations of radionuclides in liquid vault samples do not exceed the liquid concentration of the cleaned tanks."* (DOE-ID, 2005, Section 6.3, page 66)

DOE should provide the data and data quality assessment (DQA) used to support the last statement above.

Comparison of the sump samples for the two contaminated sand pads for tanks WM–185 and WM–187 to the sump samples for the other tanks with non-contaminated sand pads could provide useful information to confirm the estimated sand pad inventories presented in the 2003 Performance Assessment (PA) (DOE–ID, 2003b) and the Draft Section 3116 Determination Idaho Nuclear Technology and Engineering Center Tank Farm Facility (DOE–ID, 2005). Results of this sump sampling are not presented or discussed in documents such as the DQA Report for the Post-Decontamination Characterization of the Contents of Tank WM–185 at the Idaho Nuclear Technology and Engineering Center Tank Farm Facility (ICP, 2004e). This type of comparison is of particular importance given the potential significance of the sand pad inventories. For example, post-cleaning inventory estimations show the highest residual tank inventory (WM–182) to have a total of 2,394 Curies (Ci) compared to the contaminated sand pad total inventory estimate of 3,850 Ci (DOE–ID, 2005).

Path forward:

DOE should provide the sampling data and DQAs for the vault sumps. DOE should provide the technical basis that supports the DOE–ID (2003b) conservative estimation on the contaminated sand pad inventories and describe methods used to measure and confirm these inventories.

2 **Comment**:

Several tables used to address the waste determination inventory in the Draft 3116 Determination document (DOE–ID, 2005) need further discussion regarding unexpected results.

Basis:

Table A–12 presents a comparison of the post-decontamination estimated inventory to the conservative PA inventory (DOE–ID, 2003b). Although it is noted that the total inventory is much less than that assumed in the PA (DOE–ID, 2003b) calculation, a number of radionuclides listed in Table 1 of 10 CFR 61.55 have inventories that are larger than estimated in the "conservative" inventory of the PA. These radionuclides include Curium (Cm)-242, Np-237, Pu-239, Pu-240, Pu-241, and Pu-242. On the other hand, a greater amount of mass was removed from the tanks for important radionuclides such as Sr-90, as a result of greater than expected mass transfer into the large quantity of flushing water used to clean the tanks. No credit for this removal mechanism (desorption or dissolution) was considered in estimating the inventory for the PA; therefore, the removal is underestimated for some radionuclides.

Based on these results and a review of data provided in Table 1 (page 34) and Table A-12 (page A-68) of the waste determination, it appears that the ratios of the activities of individual radionuclides to Cs-137 (analytical or Wenzel ratios) are not constant during the cleaning process or congruent between phases. This conclusion is somewhat substantiated in the text of secondary references to a limited extent (e.g., DOE-ID, ICP-2005c) but any significant discussion in the text of the waste determination appears to be lacking. Discussion regarding the potential for preferential treatment of certain radionuclides during the cleaning process, which may be based, in part, on differences in solubility and/or partitioning of the radionuclides, would increase confidence in the results of the solids analytical results. Uncertainties associated with the solids inventory based on ORIGEN2 modeling that may under-predict undissolved fuel activities in the residual heels may also help explain some of these differences.

There is significant uncertainty in the final inventory estimate that is almost entirely based on a solid sample from tank WM-183. Analytical solids results are currently missing from the Data Quality Assessment (INEEL 2004b), although they are presented in the Engineering Design File (EDF) (Portage 2005d) for Tank WM-183. The DQA report states:

> *"Because decontamination activities reduced the volume of solids remaining in the tank to less than 15% by volume of the total sample collected, the solids portion of the samples collected were not analyzed and compared with the action levels for regulated constituents."*

Regardless of this statement, the EDF presents the solids results; however, several key radionuclides were not analyzed, e.g., H-3, C-14, Ni-63, Np-237, Pu-240, and Pu-241, while uncertain ORIGEN2 ratios were used for all radionuclides that were not analyzed in this sample.

Path forward:

Discussion regarding the apparent preferential treatment of certain radionuclides, e.g., Sr-90, and the apparent overestimation of removal for other radionuclides, e.g. isotopes of Pu, for the purposes of the PA (DOE-ID, 2003b) modeling is needed to provide confidence in the inventory results which are almost entirely based on one solid sample.

The Data Quality Assessment should be revised to include the solid analytical data to help ascertain if the data quality objectives (DQOs) presented in the sampling analysis plan (SAP) were met to enable use of the data in the decision-making process. Discussion regarding the inability to sample other tanks that contained greater solids mass is needed, e.g. WM-182 is estimated to contain almost twice as much solid mass and comparable maximum heel thickness, as well as discussion regarding the inability to sample several key radionuclides in the WM-183 solid sample.

Discussion regarding the limitations and uncertainties in the ORIGEN2 modeling particularly for estimating solids inventories in the tanks should be brought forward into the waste determination. Provide any information regarding potential mechanisms or sampling/modeling artifacts that may help explain why certain radionuclides appear to be preferentially removed during the tank cleaning process.

3 **Comment**:

Additional justification for the initial inventory for the sand pad and vault sumps appears warranted in light of the fact that no sampling data has been provided.

Basis:

ORIGEN2 modeling to estimate the inventory for WM-181 which stored SBW was used to estimate inventories for the sand pads underneath tanks WM-185 and WM-187 that stored aluminum and zirconium clad fuel reprocessing waste. Sampling data from the February 14, 1962, pre-event sampling, and samples taken from the vault during siphoning are provided in Latchum et al. (1962) but are very limited. The results of the WM-181 tank inventory estimates were calibrated to WM-185 and WM-187 analytical results just prior to the event; however, the uncertainty associated with the inventory calculation is not clear.

The performance assessment (DOE-ID, 2003b) documents calculation of initial sand pad inventories based on a diffusion model. This model was thought to be appropriate to estimate the initial inventory in the sand pad because the sand pad was saturated just prior to the event and no hydraulic gradient existed between the vault sump and sand pad. However, event documentation (Latchum et al., 1962) hypothesizes that drainage of water from the sand pad into the sump following vault sump pumping may have helped initiate the event. Additionally, according to Latchum et al., the maximum level in the annular region of the tank following the event was approximately 3.5 feet, which is well above the height of the sump (1 foot). Under these circumstances it would appear that the hydraulic head in the annulus would lead to radial, advective flow of contaminated water into the sand pad.

Additionally, even if the sand pad was saturated prior to the event and diffusion was the most important transport mechanism for mass transfer of contaminants into the sand pad, the appropriateness of a one-dimensional (1-D) diffusion model is questionable. If first-cycle extraction fluids filled the vault above the sand pad, diffusive mass transport from both above the sand pad and through the sand pad/curb interface should have been considered. An illustration showing the initial and boundary conditions for the diffusion model presented in equation 2-22 on page 2-69 of the PA is needed (DOE-ID, 2003b). There are significant inconsistencies or unexplained base case modeling assumptions that require further clarification.

As it is argued that the vaults cannot be sampled to determine the amount of residual inventory in the sand pads because they also contain diverted contamination from the valve box sumps, how was diffusion of contamination between the vault sump and the sand pad considered in the modeling of the inventory in the sand pad? Is contamination in the vault sump thought to be higher or lower than contamination in the sand pad? How is the inventory in the sumps considered in the performance assessment?

Path forward:

Provide all of the analytical data available to substantiate the initial starting concentrations for radionuclides present in the first-cycle extraction fluid that entered the vault during the event. Provide information regarding the uncertainty of the estimated starting concentrations based on ORIGEN2 modeling. Intermediate outputs from the sand pad inventory calculations are needed, such as data or figures showing the initial inventory in the sand pad after the event, and the change in inventory over time as a result of flushing events.

Additional justification for use of a one-dimensional diffusion model to estimate the base case sandpad inventory also appears warranted in light of the information reported in Latchum et al. (1962) surrounding the contamination event. DOE-ID should provide a cross-sectional diagram of the vault, tank, sandpad, curb, curb drain holes, and sump to allow review of the applicability of the diffusion model presented in the PA, along with an illustration showing the initial and boundary conditions for the 1-D diffusion model. Full explanation regarding the estimated water-level in the vault and saturation of the sand pad prior to, during, and immediately following the event should be provided. Finally, justification regarding the lack of consideration of vault sump contamination from valve box and piping encasement drainage is necessary.

4 **Comment**:

Sensitivity of model results to choice of sand pad K_d values was not discussed.

Basis:

For sorption coefficients in the sand pad, which affect both inventory and transport calculations, DOE used default sand soil values from Sheppard and Thibault (1990). No justification for this choice, nor discussion of uncertainties, was provided. Conservatism is not simply defined for these choices; for example, a lower K_d is conservative for transport but not for inventory. In the absence of site-specific data, use of default literature values needs to be justified. Sorption characteristics of the relatively clean sand likely used for the pad may differ substantially from the sand solids included in the Sheppard and Thibault compilation, which may contain up to 30 percent non-sand particles. Thus, the sand pad (if it is indeed made from clean, well-sorted sand) may have lower K_d values than sand soils.

Path Forward:

Technical bases should be provided that will build confidence in the calculation of the sand pad inventory used in the performance assessment (PA), as this inventory is estimated and does not use sampling data to build confidence that a bounding inventory has been used. Provide a discussion justifying the choice of sand pad K_d values, taking into consideration the competing conservatisms inherent in the two model uses of the parameters.

5 **Comment**:

Additional information regarding Criterion 2 evaluation is needed.

Additional information regarding analysis of tank cleaning effectiveness, estimation of residual solid volume, and remaining cleaning activities for tanks, vault sumps, piping encasements, valve boxes, and process waste lines is needed.

Basis:

Criterion 2 of the National Defense Authorization Act (NDAA), *"has had highly radioactive radionuclides removed to the maximum extent practical*," is an essential part of the NDAA. The criteria used to demonstrate removal of highly radioactive radionuclides to the maximum extent practical, taking into consideration the costs and benefits to public health and safety and the environment, plays a key role in the waste incidental to reprocessing determination process. Section 5.2 of the draft Section 3116 determination, page 50-51, defines the maximum extent practical approach as "*not removal to the extent 'practical' or theoretically 'possible'.*" Instead, it will be based on "exercising expert judgement". Expert judgement is often necessary and has an important role in waste determination processes. However, the evaluation of Criterion 2 should also include appropriate analyses that consider available waste removal technologies; technology selection, implementation, and removal effectiveness; and consideration of the costs and benefits of attempting additional removal of highly radioactive radionuclides.

An exponential decrease in concentration in the waste stream has been used to demonstrate that Criterion 2 of the NDAA has been met; however, achievement of an asymptote in the effluent concentration doesn't necessarily show that continued cleaning activities would be ineffectual. The waste determination states the following:

> *"When radiation levels decrease to the lowest value and remains [sic] constant, cleaning is stopped and the tanks are inspected. If visual inspection via a remote-controlled camera confirms that the tank has been cleaned to the extent practical [sic], then samples are collected and analyzed to verify performance objectives are met."* (DOE-ID, 2005, page 25)

There may be areas of the tanks where additional solids removal can be accomplished but the waste stream detector is only indicating diluted aqueous activities in areas of the tank that have already been cleaned. As stated in the waste determination, use of video inspection during cleaning, as well as observance of a plateau in the activity in the rinse water, needs to be used in conjunction to determine when cleaning to the maximum extent practical with the current technology has been achieved. However, the statement above is rather ambiguous, and the actual process used to determine that cleaning has progressed to the "maximum extent practical" is not clear. It is not clear how verification that performance objectives have been met via sampling is accomplished (e.g., are action levels set and if so, what are these action levels based on), or if sampling and cleaning is an iterative process. If interim sampling data exists, this data should be provided to help estimate removal efficiencies. Nonetheless, achievement of performance objectives, in itself, may not obviate the need for further cleaning.

Review of the kriged surfaces in the EDFs for each tank (Portage, 2005a; Portage, 2005b; Portage, 2005c; Portage, 2005d; Portage, 2005e; Portage, 2005f; Portage, 2005g), reveals various patterns of residual contamination in cleaned tanks. The surface for WM-182 was mistakenly included in the EDF for WM-181 (Portage, 2005b). Tanks that did not contain cooling coils appear to have less residual contamination. Some tanks appear to have a ring around the outside of the tank (e.g. WM-185), while most tanks have certain areas where cleaning appears to be less effective. Please comment on the difficulty or inability to slurry remaining solid material to jet and airlift pumps in tanks such as WM-182, which contains significantly higher residual radioactivity than other cleaned tanks.

The modeling assumptions related to the estimation of total solids volume remaining in the tanks requires further clarification. The uncertainty associated with the estimated solid volume needs to be quantified, e.g. how accurate is visual examination of reflective surfaces and reference points to solids levels in determining the measurements on this scale? More detailed information regarding the interpolation approach, e.g. grid spacing and density of reference points, is needed. Tanks with no cooling coils have a much larger density of interpolation points in some cases. Every EDF states that cooling coil brackets were used as reference points, although not all tanks have cooling coils. Please clarify these statements.

In Section 5.2.1, page 54, and Table 6 (DOE-ID, 2005), DOE uses the total production of radionuclides generated at INTEC during its operations to determine the percentages of radionuclides removed from the TFF. While it is factual that DOE did remove much of the key radioactivity from the waste stream derived from the processing of spent nuclear fuel, use of these percentages may exaggerate the effectiveness of treatment and cleaning activities. Comparing radionuclide activities present in the tanks prior to cleaning versus the activity remaining after cleaning would be useful.

Remaining cleaning activities for piping encasements, valve boxes, and vault sumps is not clear. Information regarding decontamination fluids to be used and the sequential cleaning and sampling steps left in the closure process is needed.

Provide the basis for the assumption that remaining tanks, particularly tank WM-187 that was used to collect SBW waste from other tanks undergoing cleaning operations, will be cleaned as effectively as the first seven tanks. The estimated costs associated with final clean-up activities should be provided.

DOE-ID should explicitly state whether any new technologies to clean the sand pads or tanks have been identified since the last waste determination. Provide any additional information regarding the costs of deployment of new technologies.

Path Forward:

Establish criteria that will be consistently implemented throughout the cleaning process for tanks, vaults, associated piping, and ancillary equipment. Although the final inventory status for each component after cleaning can be different, the approach should be consistent in terms of the cost/benefit analysis and the criteria employed in determining that radionuclides have, indeed, been removed to the maximum extent practical. Provide more specific details regarding the iterative process used to evaluate effectiveness of the current cleaning technology, including use of visual tools, sampling, estimation of radionuclide removal efficiencies, and any action levels used in the decision-making process used to terminate cleaning operations. Additionally, provide an evaluation of the costs and benefits of attempting additional removal of highly radioactive radionuclides.

Provide discussion regarding cleaning difficulties due to geometry or other physical limitations that make it more or less difficult to clean individual tanks. Discuss any changes in operational parameters during the cleaning process to help facilitate heel removal.

Provide the surface plots for WM-181.

Provide clarifying information regarding the data input and interpolation parameters for kriging; and the uncertainty associated with the solid surfaces used to estimate the residual solid volume remaining in the tanks.

Provide tables of percent removal based on inventory of heel prior to cleaning and after cleaning.

Discuss any remaining cleaning and sampling activities for piping encasements, valve boxes, and vault sumps. Provide additional justification regarding the expected removal efficiencies for remaining tanks to be cleaned and estimated costs for final cleaning.

Discuss the evaluation of any new technologies to remove waste from the sand pads or tanks. Please discuss whether any technologies were considered to stabilize the sand pads without removing the sand, e.g., in-situ grout injection?

6 **Comment**:

The conceptualized model showing the evolution of the vault [DOE–ID, 2003b, Figure E–1(a) and associated text depicts the outer concrete vault as uncracked in the initial phase, but shows the presence of “shrinkage cracks” on the vault and grout after the grout pour [DOE–ID, 2003b, Figure E–1(b)]. Information would be helpful regarding how the conceptual model accounts for the formation of voids and cracks due to the thermal regime resulting from the heat of hydration of the grout pour and due to the physical coupling between concrete and grout.

Basis:

The evolution of concrete and grout properties depends partly on the extent of physical coupling between two materials of different compositions and partly on the thermal regime initiated by heat of hydration of the grout pour. The situation is further complicated by the fact that the concrete is subject to restraint, by reinforcement, whereas the grout is apparently unrestrained. The evolution of concrete and grout properties occurring at this stage may have an important influence on the final grout and concrete properties. Laboratory simulations do not simulate the coupling between materials at full scale and properties measured on these simulants may not scale up.

Path forward:

Provide a technical basis for the anticipated thermal and mechanical history of the various cementitious materials. Provide any available data on the isothermal shrinkage of the two main material types (concrete and grout) and assess the additional impact of the thermal excursion arising in the course of emplacing the grout and its subsequent hardening. Explain how the dimensional changes would translate to the incidence of voids and cracking and the five stages illustrated in DOE–ID (2003b, Figure E–1).

7 **Comment**:

In the conceptual model, degradation is assumed to occur from the outside and propagate inwards, and a homogeneous degradation front is modeled (DOE–ID, 2003b). This conceptual model may not be appropriate if the transport of aggressive agents is through cracks or if degradation occurs from internal factors (e.g., by alkali-aggregate reaction).

Basis:

As indicated in Comment 6, there is a potential for cracking that apparently has not been evaluated. If cracking occurs, and especially if cracking is confined to a relatively few but wide cracks, each crack may serve as a rapid conduit for migration of water deep into the grout mass. Cracks may eventually dominate the rate at which degradation occurs, especially in grout which is unrestrained.

Path Forward:

How have alternative conceptual models of degradation that consider the effects of the emplacement of grout and the resulting thermal regime, as well as the intrinsic dimensional stability (i.e., under isothermal conditions) of the grout as a function of time and hydration of its components been considered? If these alternative conceptual models have not been considered, explain how the analysis presented in Appendix E is expected to bound the potential results of these alternative conceptual models?

8 **Comment**:

DOE has not fully justified its choice of the surface rinsing release option in DUST–MS.

Basis:

The surface rinsing model uses an equilibrium sorption relationship to calculate pore-water radionuclide concentrations in the grout and sand pad sources. Among alternative release models is one that assumes that radionuclide water concentrations are controlled by solubility. It is not always clear which type of model is most appropriate or conservative. Typically both models are considered, and the model that provides the most conservative result is used. On page 2-57 of DOE–ID (2003b), DOE makes clear that the solubility limit model was not used, but also refers to comparisons between tank grout pore-water concentrations based on solubility and those based on sorption coefficients. The details of this comparison are not provided, but are necessary for evaluating the choice of release model. Discussions of release model choice in Section 7.7 of DOE–ID (2003b) do not include a comparison with the solubility model.

Path Forward:

Provide a discussion and tabulation of the pore-water concentration comparison (DOE–ID, 2003b, Section 2.1.6.3, p. 2-57) and justification for the model selected.

9 **Comment**:

Flooding at the TFF could increase the downward mobility of subsurface contamination, potentially influencing flow regimes, transport pathways, transport times, and contaminant concentrations. Additional technical basis is needed to support the flooding analysis.

Basis:

Figure 2-18 shows the areal coverage of a Probable Maximum Flood at INL (DOE–ID, 2003b). It is difficult to review Figure 2-18 because its resolution is low. The figure would be more useful if shown at a higher resolution. The location of Idaho Nuclear Technology and Engineering Center is not superimposed on the inundation map shown in Figure 2-18, yet its location apparently near a margin of the flooded area is supposed to support the flooding analysis.

It is not clear in the performance assessment (DOE–ID, 2003b) what assumptions are made regarding dam maintenance, effectiveness of the INL spreading areas and flood diversion facility, and landscape evolution over the period of performance. Such assumptions would be relevant to severity and implications of flooding at the TFF.

Section 2.1.5.3.6 pertains to Potential Dam Failures. DOE–ID (2003b) in addressing the study performed by Van Hafften, Koslow, and Naretto (1984) for the flooding analysis of the New Production Reactor site, includes the following text: "The flood hydrograph for the base-case discharge shows a peak flow that lasts only 7.5 hours (Figure 2-19). Since the INTEC facility is located at the boundaries of the flood, the area would be inundated for a lesser period of time" (p. 2-38). The information conveyed in the second sentence called out above does not necessarily follow from the information conveyed in the first sentence: The duration of peak discharge does not necessarily imply how long water may pond at the surface of the TFF or elsewhere (see also Section 7.4.2, page 7-26).

The maximum depth to which ponded flood water is expected to infiltrate into the unsaturated zone below the TFF requires a technical basis (DOE–ID, 2003b, p. 2-39). The concept of a "wetting front," which is common to conceptual models of flow through sedimentary porous media, may not be appropriate for the fractured basalt portions of the unsaturated zone below the TFF (e.g., Faybishenko, et al., 2000).

For the Large Scale Infiltration Test conducted at INL, Newman and Dunnivant (1995) were able to model only 10 of 26 breakthrough curves using a one-dimensional transport model, which suggests use of an inadequate method (Wood and Faybishenko, 2001).

Path Forward:

DOE–ID should provide a higher resolution figure of the relevant portions of Figure 2-18. DOE–ID should also show the location of INTEC superimposed on the higher-resolution inundation map.

DOE–ID should provide information regarding their assumptions for landscape evolution, which are relevant to the severity and implications of flooding at the TFF. This information may include assumptions by the Army Corps of Engineers regarding the useful lifetime of Mackay Dam, and whether Mackay Dam is assumed to be dredged repeatedly so that it does not fill with sediment. This information may include an analysis of flooding scenarios and effect on dose if Mackay Dam was no longer maintained starting at t = 500 years (not a one-time dam failure, but rather the dam is assumed out-of-commission over the long term). DOE–ID should provide its assumptions and technical bases regarding whether the spreading areas at INL will not fill with sediment such that capacity is reduced. DOE–ID should provide an estimate regarding the longevity of the effectiveness of the flood diversion facility in the absence of maintenance at the end of the period of institutional control.

DOE–ID should provide the text on page 2-35 of the DOE–ID (2003b) that is currently hidden behind a figure.

DOE–ID should provide information that demonstrates that the impacts of the probable maximum flooding condition on the INTEC facility have been bounded. DOE–ID should provide the analysis by Dunnivant, et al. (1998) on which they, in part, base their expectation of minimal impact. DOE–ID should address the potential for higher levels of contamination to persist for a longer period of time in the flooding scenario. DOE–ID should provide an analysis that clearly demonstrates the expected amount of time water might pond at the surface of the TFF after a dam failure or probable maximum flood.

10 **Comment**:

Based on review of new information, it is not clear that the current two-dimensional unsaturated zone groundwater model (DOE–ID, 2003b, p. 3-39) is adequate for estimating the all pathways dose to a member of the public, and additional information is needed to determine if results from the two-dimensional model are conservative.

Basis:

The three-dimensional geometry of the *intra*-basalt fracture system and *inter*-basalt rubble zones is important (Faybishenko et al., 2001), but the geometry is not accounted for in the DOE–ID two-dimensional groundwater model. At the scale of the TFF groundwater model, fracture patterns likely are not isotropic in map view. Basalt flows typically have a lobate distribution 20–60-m wide and up to 1,000-m long (Sorenson, et al., 1996). Detailed vertical stratigraphy in two-dimensions cannot account for tortuous flow paths expected in three dimensions due to the presence of basalt flow fingers or lenticulae, where upper margins of flows are rubbly and fractured (due to fast cooling) whereas flow interiors and bottoms are massive and less fractured (due to slow cooling). Rubble zones associated with individual flow tops may also conduct lateral groundwater flow, and the decreasing fracture spacing with depth into a single basalt flow leads to fewer flow paths and potential funneling—a process observed to continue with increasing depth as water leaves shallow basalt flows and moves into deeper basalt flows (Faybishenko, et al., 2000). Physical fast transport pathways could occur in the study area because of heterogeneity in active lateral basalt rubble pathways and lateral sedimentary interbed pathways, and heterogeneity in active subvertical fracture pathways. Multiple peaks in breakthrough curves during field tests at INL suggest water follows a number of different transport pathways (e.g., Jones, et al., 2004 and LSIT results). While it may be correct in part to suggest that lack of transverse dispersion in a two-dimensional model could contribute to achieving a conservative dose estimate, adopting this methodology neglects that a two-dimensional model may be prone to artificially "trapping" water and radionuclide contaminants because the connectivity of actively conducting portions of the model domain is too low in the absence of the third dimension (Doughty, 2000). To the extent that radionuclide-bearing water may be artificially trapped in the model domain, this methodology is nonconservative. Capturing a large level of detail in two dimensions may be of less value than capturing an appropriate level of detail in three dimensions.

Wood and Faybishenko (2001) say that at INL "without an overall understanding of the geometry and physics of flow at the macro-scale, we can describe what is seen, but cannot use this information for making further predictions, and therefore cannot make meaningful assessments of contaminant transport."

Previous recommendations of Lawrence Berkeley National Laboratory and INL staff (Faybishenko, et al., 2001; Doughty 2000) for achieving realism in vadose zone flow and transport models of INL include simulating the structured, nonrandom geometry of basalt fracture patterns using deterministic fracture models in addition to simulating flow in the system using stochastic hydrologic parameter models.

Path Forward:

DOE–ID should provide any available information on the geometry of the fracture system in the unsaturated zone downgradient from the TFF. DOE–ID should provide any available information on the anisotropy of basalt flows in the TFF vicinity, and to what degree the long lateral axis of basalt flows are oriented obliquely to hydraulic gradients at the scale of the TFF groundwater model. Information and data from other sites at INL (e.g., Radioactive Waste Management Complex, Large Scale Infiltration Test, and Vadose Zone Research Park) may provide a supporting basis for the selected modeling approach. DOE–ID should provide information that demonstrates that they are either realistic or conservative in their dose estimates for the groundwater pathway.

11 **Comment**:

Unsaturated zone hydrologic properties of massive-to-fractured basalts, basaltic rubble zones, and sedimentary interbeds are highly uncertain, as are the initial conditions for radionuclide release, and the nature of the flow history prior to initial release. These sources of uncertainty are not reflected in the deterministic model results. In the current model, significant emphasis is placed on the sedimentary interbeds but, during the Large Scale Infiltration Test, intra-basalt flow, column-bounding vertical fractures were shown to contribute significantly to the vertical permeability of the vadose zone at INL (Faybishenko, et al., 2000).

Basis:

The unsaturated zone parameter values, constitutive relationships, and the treatment method for vadose zone air in the flow and transport model will affect unsaturated zone flow and radionuclide transport model estimates. Field-scale evidence from tests at INL or just outside the INL boundary at the Box Canyon analog site suggests that flow patterns and zones of elevated saturation change over relatively short time-scales, and that they are sensitive not just to lithology, but also to initial conditions and the nature of previous episodes of wetting and drying (Faybishenko, et al., 2000).

Parameter ranges and constitutive relationships used in the flow model should be clear to the reviewer, and parameter uncertainty should be evaluated to determine a defensible unsaturated zone parameter set. Information, data, and model results from other sites at INL (e.g., Radioactive Waste Management Complex, Large Scale Infiltration Test, Vadose Zone Research Park) may provide a supporting basis for parameter values used in the tank farm groundwater model.

Path Forward:

DOE should provide the lithologic analysis of Anderson, et al. (1999). DOE–ID should clearly identify how they treat the gaseous phase in their PORFLOW model (i.e., is air treated as stagnant, or is the groundwater model for the TFF evaluation truly a multiphase flow simulation?). DOE–ID should evaluate the significance of individual components of a recommended multi-geological-component conceptual model (Faybishenko et al., 2000) for unsaturated zone flow at INL, including (1) fracture-to-matrix diffusion, (2) vesicular basalt-to-massive basalt diffusion, (3) preferential flow through conductive fractures and the effect of flow funneling, (4) vesicular basalt-to-nonconductive fracture diffusion, (5) conductive fracture-to-vesicular basalt advection and diffusion, (6) lateral flow and advective transport in the central fracture zone, (7) lateral flow and advective transport in the rubble zone, and (8) vertical flow and advective transport into underlying basalt flows (Faybishenko et al., 2000). Justification should be provided for not including significant components in modeling analyses for estimating groundwater dose to the public.

12 **Comment**:

Overall, information pertaining to geologic controls on hydrology for the TFF vicinity are lacking from DOE–ID (2003b) (see Comments 10 and 11). Information provided does not address whether there is a known southerly dip of the sedimentary interbeds, consistent with the selected model domain. The regional potentiometric map is not shown, and while there is likely a dominant direction of basalt flow fingers or lenticulae, their orientation is not addressed in the performance assessment.

Basis:

An important geologic control on unsaturated zone flow is the dominant dip direction of sedimentary interbeds. While flow in the aquifer may be to the south-southwest, it is unclear whether perched water flow along sedimentary interbeds would also preferentially flow to the south.

The regional potentiometric map is a source of groundwater model boundary conditions, but a map illustrating this information is not provided. Without this information the reviewer must rely on statements that saturated groundwater flow is to the south-southwest for a model that predicts flow only in a due south direction. The dominant orientation of basalt fingers or lenticulae should also be noted on the map.

Path Forward:

DOE–ID should provide reports cited on page 2-47. These documents include Walker (1960), Ackerman (1991), and Anderson, et al. (1999). DOE–ID should provide information that addresses the open question of whether the sedimentary interbeds dip in a known direction, and if so, are they known to have a dominant southerly dip consistent with the modeled domain? If the sedimentary interbeds are known to have a dominant dip in a direction that is not consistent with the modeled domain, DOE–ID should address the implications of modeling water flow in a direction that is oblique to the dominant dip direction.

DOE-ID should provide a map of the regional potentiometric surface, and an indication of the dominant direction of basalt fingers or lenticulae.

13 **Comment**:

Existing sampling data for radionuclides currently contaminating the subsurface of the TFF are not provided. This data may provide useful information regarding the accuracy or degree of conservatism of the flow and transport model.

Basis:

Existing soil contamination (from piping leakage) has resulted in the transport of Tc from the tank farm source area to the saturated zone underneath the TFF in less than fifty years. It is not clear that this information is consistent with unsaturated zone flow and transport modeling predictions.

The initial radionuclide concentrations are set to zero at the model boundaries and in the model domain, even though there is known contamination of the subsurface at the TFF (DOE–ID, 2003b, p. 3-36). This treatment is said to be based on existing sampling data that shows existing contamination of the site is minimal, especially for future estimates.

Path Forward:

DOE–ID should provide existing sampling data for radionuclides currently contaminating the subsurface at the TFF and information regarding saturated groundwater contamination within the model domain. DOE-ID should discuss to what extent monitoring data in the vicinity of the TFF corroborates the modeling predictions presented in the 2003 PA. Contaminant releases to the aquifer from the injection well, WAG-3 soils, and TRA warm waste ponds were evaluated in the WAG-3 Remedial Investigation/Feasibility Study (RI/FS) (Rodriquez et al., 1997). DOE-ID should provide this reference.

14 **Comment**:

It is not clear that the unsaturated zone flow and transport model accounts for potentially limited solute-rock interaction during flow along basalt fractures.

The potential for colloid-facilitated release and transport was not addressed.

Basis:

As mentioned in Portage (2005j, p. 12), a previous modeling effort at INL employed zero sorption coefficients for all contaminants in the unsaturated zone because of the relatively rapid velocity of water in vertical fractures in the basalt (Rodriguez, et al., 1997). In subsequent discussions, DOE does not justify its use of non-zero sorption coefficients in the unsaturated zone.

Colloids may enhance release and transport of otherwise relatively immobile radioelements such as plutonium. While it is not clear that colloids could be important for system performance, no justification was provided for neglecting colloidal effects.

Path Forward:

DOE should explicitly justify the use, in its compliance calculations, of non-zero sorption coefficients in unsaturated basalt in light of the conclusions of Rodriguez, et al. (1997). If retardation in unsaturated basalts can be shown to have a negligible effect on performance, further justification is not needed.

Provide a discussion supporting neglect of colloidal effects on radionuclide release and transport.

15 **Comment**:

It is not clear to what degree existing and more recent sorption data relevant to INL were used in developing and evaluating values used for the performance assessment.

Basis:

Site-specific sorption data used in developing and justifying performance assessment parameters for carbon, iodine, strontium, and technetium are limited to only two references (Del Debbio and Thomas, 1989; Rodriguez, et al., 1997). The K_d evaluation (Portage, 2005j) mentions a field-based study that provided K_d information (Beasley, et al., 1998), but does not quote values nor use this study in developing performance assessment parameters. In addition, it is notable that the most recent sorption reference in Portage (2005j) is from 2000. It is possible that more recent information may be available; for example, Fjeld, et al. (2001) present data on strontium sorption in basalt and sediment column experiments.

Path Forward:

Confirm that no available sorption data relevant to INL have been neglected in developing and supporting performance assessment parameters.

Also, please provide the following references:

Del Debbio, J. A. and T. R. Thomas. “Transport Properties of Radionuclides and Hazardous Chemical Species in Soils at the Idaho Chemical Processing Plant.” WINCO–1068. Idaho Falls, Idaho: Westinghouse Idaho Nuclear Company, Inc. 1989.

Rodriguez, R.R., A.L. Shafer, J. McCarthy, P. Martian, D.E. Burns, D.E. Raunig, N.A. Burch, and R.L. VanHorn. “Comprehensive RI/FS for the Idaho Chemical Processing Plant OU 3-13 at the INEEL—Part A, RI/BRA Report (Final).” DOE/ID–10534, Binders 1–3. Idaho Falls, Idaho: DOE. 1997.

16 **Comment:**

The assumptions made in the performance assessment for the acute and chronic well intruder scenario should be analyzed quantitatively in a sensitivity study.

Basis:

The assumptions regarding the depth and area of contamination significantly impacts the resulting dose for the acute or chronic well intruder scenarios. The assumption that the waste is spread out over a large area (2200 square meters) effects the thickness of contamination, the proximity of the receptor to the contamination, and the resulting doses from the external dose pathway. The basis for the assumed 6" diameter (vs. 8" diameter) well for the chronic well intruder scenario, as well as the assumed area of contamination significantly impacts the concentrations and dose from the external dose pathway.

Equation 5-19 in the PA appears to be incorrect. Furthermore, uncertainty in the inventory for the sand pads should also be taken into consideration in these calculations.

Path Forward:

Please evaluate the sensitivity of modeling results to the thickness (based on well radius and area assumption) and concentration (based on well radius, area and mixing depth assumption) of the contaminated zone for the acute and chronic well intruder scenarios, respectively. Consideration of the uncertainty associated with the sand pad inventory should also be considered.

Correct Equation 5-19 and verify whether this equation was a typographical error, or if the equation was actually used in the PA as presented.

17 **Comment:**

DOE-ID needs to determine if the final end-state of residual contamination in grouted tanks, vaults, and auxiliary equipment at the TFF is Class C or greater as defined in 10 CFR 61.55.

Basis:

The Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005

(NDAA) provides criteria for determining whether certain waste resulting from the reprocessing of spent nuclear fuel is not high-level waste (HLW). Criteria 3(A) and 3(B) of Section 3116(a) of the NDAA require that the waste be disposed of in compliance with the performance objectives contained in NRC regulations at 10 CFR 61, Subpart C. The applicability of either 3(A) or 3(B) is dependent upon whether the waste exceeds Class C concentration limits, thus the classification of waste residuals must be determined in order to apply the NDAA criteria.

Path Forward:

DOE-ID should consult the interim concentration averaging guidance (70 FR 74846) for additional information regarding acceptable methods of estimating residual concentrations in TFF tanks, vaults, and auxiliary equipment. DOE-ID needs to specify the class of residual waste at the TFF, as defined in 10 CFR 61.55. Assumptions used in the calculation of waste concentrations should be clearly stated and justification for these assumptions should be provided.

CLARIFYING/INFORMATION REQUESTS:

1 Information on verification of the FORTRAN sand pad inventory code was not provided. Please provide any information on verification of the FORTRAN code used to calculate sand pad radionuclide inventories.

2 Clarify whether localized production of carbon dioxide from microbial activity in the subsurface has been considered in estimating the impacts of carbonation on concrete degradation.

3 The performance assessment calculations assumed that reducing conditions would prevail and used K_d values appropriate for reduced environments. No technical basis was provided for this assumption except a statement [page 7-11 of the performance assessment (DOE–ID, 2003b)] that "The concrete is expected to exhibit strong reducing conditions (Eh from !300 to !500 mV) as do most concrete systems." This statement is incorrect. Measured values of Eh of Portland cement lie in the range +0 to +100 mV, and the redox potential is not well buffered and could easily change by the introduction of other electroactive species (Atkins and Glasser, 1992). Slag-rich cement blends, however, tend to have more reduced environments due to reduced sulfur species [from sulfides released on hydration of the blast furnace slag (BFS) glass], giving rise to strongly reducing conditions. For example, a mean Eh value of !305 mV was obtained on slag-cements (85-percent BFS) aged between 1 and 10 months (Atkins and Glasser, 1992). Reduced forms of iron and manganese (present in BFS) are unlikely to play a major role in establishing the redox level, because of their relative insolubility at high pH.

Although the performance assessment also states "the closure system will also consist of a mix of concrete and fly ash, slag, or other substances to ensure reducing conditions in the grout," (page 7-11), it would be more appropriate to explicitly state that slag will be added to the concrete and grout to ensure the establishment of a reduced environment and mitigate the release of electroactive radionuclides, such as technetium (Tc)-99. Please clarify the percentage of slag to be used in the concrete or grout mixture to allow evaluation of whether a reduced environment will be supported.

4 The performance assessment states, "the environment surrounding the vault is not aggressive and recent photos of the vault walls (i.e., 40- to 50-year old concrete) show no evidence of cracking" (DOE–ID, 2003b, p. E–20). The significance of this observation is not clear. When carbonation reaches the depth of the steel, accelerated steel corrosion may occur. The performance assessment does not indicate the thickness of the cover concrete, but it is likely carbonation has not yet reached the embedded steel. If that thickness were known, perhaps from the original specification, it could provide a maximum rate of carbonation that would be useful in bounding the calculations discussed in DOE-ID (2003b). Provide a more transparent application of the qualitative observation.

5 In DOE–ID (2003b, Appendix E), the point at which degradation of the vault to rubble occurs is taken as "when 50 percent of the reinforcement steel corrodes." It is not clear from the document why this value was selected and how it is defined, e.g., 50-percent

loss of cross section or 50-percent loss of total mass. Clarify what is meant by 50 percent corrosion of the reinforcement steel.

Depending on the definition, the calculation may be sensitive to the physical dimensions of the rebars. On page E–22 (DOE–ID, 2003b, Appendix E), it is stated the reinforcement steel has a diameter of 0.25 inch. However, no basis is provided for this value. In a structure with the physical size of the tanks, larger diameter steel reinforcement would normally be specified. Because a constant rate of corrosion is applied in terms of thickness corroded per year, the calculation of the time required to reach 50 percent will be sensitive to the choice of rebar dimensions. Provide a reference for the reinforcement steel diameter.

6 DOE–ID (2003b, p. E–21) states that the value for the bulk concentration of $Ca(OH)_2$ in solid concrete used in Eq. E–14 is 27.5 kmol/m^3. This value appears to be too high, overestimating the reserves of $Ca(OH)_2$, and resulting in a lower rate of carbonation (dX/dt, Eq. E–14). Assuming a relatively cement-rich concrete is used, containing 400 kg of cement m^3 of concrete, modern Portland cement would hydrate to give about 20-percent $Ca(OH)_2$. That is, the concrete will contain about 80-kg $Ca(OH)_2/m^3$.

For a volume of 1 m^3, this is equivalent to

$$\frac{80{,}000 \text{ grams}}{74 \text{ grams / mole}} = 1{,}081 \text{ moles } Ca(OH)_2$$

The difference between 27.5 kmol/m^3 and 1.08 kmol/m^3 could be significant in terms of predicting the rate at which carbonation occurs.

Elsewhere in the text (DOE–ID, 2003b, p. E–22) a much lower value of $Ca(OH)_2$ content, 1.875 kmol/m^3, is used for calculating $Ca(OH)_2$ leaching. Appendix E assumes that all the calcium is available. However, as calcium leaching progresses, the pH declines. For practical purposes, and using a conservative approach, only that fraction of calcium leached at high pH (above 12.4 at 20 EC) should be considered. Because much of the calcium is present in phases other than $Ca(OH)_2$, it would be preferable to count only the contribution from $Ca(OH)_2$. Hence the value 1.875 kmol/m^3 is too high if a conservative approach is to be maintained. Please clarify the discrepancies and technical basis for the assumed concentrations.

7 Figure 2-20 (DOE–ID, 2003b) presents relative permeability and moisture characteristic curves based on data presented in Rodriguez, et al. (1997). It is unclear whether the constitutive relationships in Figure 2-20 was used for the TFF performance assessment (2003b). If the relationships in Figure 2-20 were used in DOE–ID (2003b), small pores and absorbed water films are neglected (i.e., there is no residual water saturation using this relationship) and water drainage begins instantaneously with capillary pressure decrease (i.e., there is no air-entry pressure using this relationship). It also appears that only drainage (but not imbibition) is modeled. If a relationship by Magnuson (1995) was used, as cited on page 7-32 (DOE–ID, 2003b), it is unclear how the relationship by Magnuson differs from the one shown in Figure 2-20. Please clarify what relationship was used in the modeling.

8 Final calibrated hydraulic conductivities for major geologic layers should be provided, e.g., upper and lower perched zones.

9 Section 2.1.5.3.9 pertains to the lower perched zone (DOE–ID, 2003b). The nature and extent of the uncertainty in characterizing the lower perched zone is not clear. The statement made on page 2-44 “these data contain a high degree of uncertainty, since they consist of a combination of original driller’s logs (some dating back 40 years), geophysical borehole logs, and monitoring wells that are completed in this zone,” (DOE–ID, 2003b) requires more information. DOE–ID should provide more specific information regarding the source and implications of the subject uncertainty.

10 The base of the groundwater model is set as a no flow boundary, partially based upon an assumption of “nonexistent future pumping of the aquifer for water supply.” The appropriateness of this assumption is unclear. A more appropriate assumption might be that future pumping of the aquifer is expected to impart negligible impact on horizontal flow within the model domain, substantiated by a simple analysis given future expectations for water supply, pumping rates, and expected spatial intervals of water supply wells. DOE-ID should provide the technical basis for the no-flow boundary.

11 DOE–ID should provide a map showing the spatial relationship of the new Vadose Zone Research Park percolation ponds to the TFF, and indicate if their will be any influence from these percolation ponds on the formation of perched water in the study area and/or on contaminant flow and transport predictions. DOE–ID should clarify which figure in the PA shows the final distribution of perched water, as estimated via calibration, or provide this figure if it is not already included.

12 With regard to the piping inventory, the waste determination states: *"The pipe encasements are similar to valve boxes in that they do not normally contain process solutions; they are designed so that any leakage into the pipe encasements would drain to sumps, which is then jetted to tanks; and the contamination in the secondary containment is addressed by using the safety factor of 500 established for the piping inventory. The safety factor of 500 is described in Subsection 2.3.1."* DOE should provide a better rationale for the use of this "500" factor. Subsection 2.3.1 only indicates the factor is used to create a conservative result.

13 Figure 3-14 shows the conceptual model for the DUST-MS release modeling. The figure implies that release from the sand pad does not occur until the grouted tank fails at 500 years, i.e., there is a hyrdraulic barrier limiting infiltration to the sand pad until after 500 years. Please confirm that this conceptual model is for the tank release only.

14 The DUST-MS release rates presented in Appendix F and Chapter 4 of the PA are significantly different (DOE-ID 2003b). For example, the Sr-90 peak release rate in Figure F-11 on page F-8 approaches 0.1 Ci/yr, while the release rate Figure 4-1 on page 4-2 approaches 0.001 Ci/yr. Discrepancies of similar magnitude exist for Tc and I as well. Please explain the differences in the release rates.

15 Please clarify how the DUST-MS release rates were incorporated into the PORFLOW model. Detail any averaging assumptions in space and time of the source input that

were necessary to facilitate differences in model scale, e.g., grid and time step size of the PORFLOW model.

16 On page 3-54 and 3-55, DOE-ID discusses calibration of the transport model using tritium data from the percolation ponds (Orr and Cecil 1991). It is not clear that the most appropriate comparisons were made with the data. Please clarify the extent of tritium contamination versus the modeled predicted extent of contamination (only a comparison of concentration was made). Please provide the Orr and Cecil reference.

17 In Portage (2005j), DOE justifies its choices of sorption coefficient values for the compliance or “conservative” case and for the other cases (“worst-case,” “realistic,” and “best”) used in the sensitivity analysis presented in DOE–ID (2003b). The values are based on a combination of literature and site-specific data. As discussed below in more detail for each radioelement, there are three common problems in the Portage (2005j) discussions: (i) values for the “realistic” and “best” cases are identical and the label “realistic” is typically used for the upper bound values, rather than being based on anything demonstrably realistic; (ii) “conservative” values are sometimes arbitrarily chosen; and (iii) “worst-case" values typically should be more accurately termed “conservative.” This discussion will focus on the “conservative” case values, because this case is used for comparing with performance objectives. Note that, with the exception of Tc, the grout discussions apply also to concrete, for which DOE uses the same K_ds.

Strontium—For basalt, site-specific laboratory studies yielded K_ds in the range 1.1 to 3.4 mL/g (Del Debbio and Thomas, 1989; Porro, et al., 2000), yet the “conservative” performance assessment value is 6 mL/g (chosen as a midpoint between low and high values). The “worst-case” value of 1 mL/g is more appropriately termed “conservative,” because it bounds the range of laboratory data. Two INL-specific studies yielded Sr basalt values of 0.5 and 3.0 mL/g (Arnett, et al., 1990; Robertson, 1974); it would appear unreasonable to adopt a “conservative" K_d higher than these. For interbed sediments, the values overall appear reasonable, although the “conservative" value was arbitrarily chosen within a range (note also that the confirmatory Japan Nuclear Cycle data are for “mudstone,” which may not be generally appropriate for INL interbeds). For grout, the Bradbury and Sarott (1995) compendium recommends 1 mL/g as a conservative value, yet DOE uses this value for their “worst-case.” The “conservative” value of 3 mL/g was arbitrarily scaled between high and low values. Note that Atkinson and Nickerson (1988), cited in Bradbury and Sarott (1995), recommended 3 to 6 mL/g as a best estimate range; this terminology appears appropriate for a “realistic,” rather than “conservative,” case.

Technetium—For basalt and interbed sediments, the “conservative” K_d of 0.01 mL/g appears to be appropriate on the basis of the literature, but this choice needs to be better reconciled with the selection of 0 mL/g by Rodriguez, et al. (1997). For grout, Bradbury and Sarott (1995) recommend 1000 mL/g as a conservative value for reducing conditions. Clearly, the “worst-case” would be if Tc were oxidized, in which case Bradbury and Sarott recommend 1 mL/g. Assuming that DOE can justify maintenance of reducing conditions, an appropriate “conservative” value would be 1,000 mL/g, rather than the DOE value of 2,500 mL/g, arbitrarily scaled between 1,000 mL/g and the highest literature value of 5,000 mL/g. Note that the latter value would be more appropriately termed “best” rather than “realistic.”The value of 1 m^3/kg for technetium K_d used for the worst-case scenario in the groundwater pathway sensitivity analysis (DOE–ID, 2003b, Section 7.3) neglects the diffusion or advection of oxygen into the concrete and grout that could produce locally oxidized conditions where concrete and grout contact the external environment. These oxidized zones could allow certain radionuclides, particularly Tc-99, to be more mobile and govern radionuclide release rates.

Iodine—The basalt “conservative” K_d of 0.1 mL/g needs to be reconciled with (i) the “site value” of 0 mL/g in Portage (2005, Table 5) and (ii) the lower NEA database value of 0.05 mL/g. For interbed sediments, no basis is provided for the apparently reasonable “conservative” value of 0.1 mL/g. For grout, DOE has chosen, without justification, to use the Bradbury and Sarott (1995) conservative K_d of 2 mL/g as the “worst-case” value. The DOE “conservative” value is arbitrarily scaled between 2 mL/g and the “best” (not “realistic”) literature value of 30 mL/g; the Bradbury and Sarott (1995) conservative value would be more appropriate.

Carbon—Basalt K_ds are from Sheppard and Thibault (1990) sand soil values (based on three observations), with the lower end used for the “worst-case” and the mean used for the “conservative” case. The acknowledged uncertainty in these assignments (Portage, 2005, p. 13) would appear to suggest that the low end value be used for the conservative case. The same may be said for the sediment values; the lowest reported value for sand soil (1.7 mL/g) would appear to be an appropriate “conservative” choice. For grout, it would be useful to know if the investigators attempted to calculate a carbon K_d using the approach of Bradbury and Sarott (1995).

DOE should be judicious in choices of “conservative” K_d values. Such values, when used for the compliance demonstration, should be demonstrably bounding at the low end of reasonably expected values. In many cases, DOE has instead used such lower bounds for the “worst-case,” with little explicit basis for “conservative” values. This approach does not build confidence in the applicability of the compliance demonstration doses. DOE should consider revising the “conservative” values so that there is greater confidence that their “conservative” values reasonably and defensibly bound the expected range. Likewise, DOE should use only the label “best,” and not “realistic,” for its high-K_d cases. In only some of the cases is there a basis for terming the values “realistic.”

18 List key radionuclides for worker dose during closure and clarify if short-lived radionuclides were screened out during the analysis to identify these radionuclides, i.e., confirm whether short-lived radionuclides would contribute to worker dose during closure activities.

19 Table 3 (page 39) presents the sand pad residual inventory at closure in Ci per sand pad. The value for Cesium (Cs)-137 is presented as 2.53E-06 Ci. Based on a number of other references to this inventory, including Table 14 (page 68) of this document, and the activity for Ba137m, the Cs-137 inventory in the sand pad is expected to be on the order of 1.6E+03 Ci. The value for Pu-238 in Table 3 (page 39) of the waste determination (DOE-ID, 2005) is 5.06E-06 Ci, while the value presented in the PA (DOE-ID, 2003b) is 2.0 Ci (see Table 2-17 on page 2-73). Additional radionuclide

inventory estimates are discrepant between the waste determination and PA tables. Clarify the correct value for the sand pad inventory and confirm which values were used in the performance assessment.

20 There are several inconsistent and redundant statement in the performance assessment (DOE-ID 2003b).

- On page 2-68, it states that the center of the sand pad is 2 inches in one sentence and 3-4 inches in the next sentence.
- On page 2-70, it states that in the *"absence of data"*, 38 flushing events were assumed, while on page 2-68 it implies that records exist that suggest over 100 flushing events have occurred.
- On page 4-5, it states that source release for two tanks is loaded at the location of the southern tank, while on page 3-37 it states that releases from one tank were simulated and the concentrations doubled at downgradient locations.

Indicate which value or statement is correct.

21 Please provide the following references:

Ackerman, D.J., 1991, "Transmissivity of the Snake River Plain Aquifer at the Idaho National Engineering Laboratory Site, Idaho, USGS Water-Resources Investigations Report 91-4058, DOE/ID-22097.

Anderson, S.R., M.A. Kuntz, and L.C. Davis, 1999, "Geologic Controls of Hydraulic Conductivity in the Snake River Plain Aquifer at and near the Idaho National Engineering and Environmental Laboratory, Idaho, U.S. Geological Survey Water-Resources Investigations Report 99-4033, DOE/ID-22155.

Arnett, R.C., R.C. Martineau, and M. J. Lehto. "Preliminary Numerical Model of Radionuclide Transport in the Snake River Plain Aquifer Near the Idaho National Engineering Laboratory." EGG–WM–8820. Idaho Falls, Idaho: EG&G Idaho, Inc. 1990.

Del Debbio, J. A. and T. R. Thomas. "Transport Properties of Radionuclides and Hazardous Chemical Species in Soils at the Idaho Chemical Processing Plant." WINCO–1068. Idaho Falls, Idaho: Westinghouse Idaho Nuclear Company, Inc., 1989.

Dunnivant, F. M., et al., 1998, "Water and Radioactive Tracer Flow in a Heterogeneous Field-Scale System," Ground Water, Vol. 36, No. 6, pp. 949-958.

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Orr, B.R., and L.D. Cecil, 1991, "Hydrologic Conditions and Distribution of Selected Chemical Constituents in Water, Snake River Plain Aquifer, INEL, Idaho, 1986-1988," USGS Water-Resources Investigation Report 89-4008, DOE/ID-22078.

Robertson, J.B. “Digital Modeling of Radioactive and Chemical Waste Transport in the Snake River Plain Aquifer at the National Reactor Testing Station, Idaho.” U.S. Geological Survey Open-file Report IDO–22054. 1974.

Rodriguez, R.R., A.L. Shafer, J. McCarthy, P. Martian, D.E. Burns, D.E. Raunig, N.A. Burch, and R.L. VanHorn. “Comprehensive RI/FS for the Idaho Chemical Processing Plant OU 3-13 at the INEEL—Part A, RI/BRA Report (Final).” DOE/ID–10534, Binders 1–3. Idaho Falls, Idaho: DOE. 1997.

Sorenson, K.S., A.H. Wylie, and T.R. Wood, Test Area North Hydrogeologic Studies Test Plan: Operable Unit 1-07B, Rep.; INEL-96/0105, Idaho Natl. Eng. and Environ.Lab., Idaho Falls (1996).

Walker, E.H., 1960, “Analysis of Aquifer Tests, January 1958-June 1959, at the National Reactor Testing Station, Idaho,” Atomic Energy Commission.

REFERENCES:

Arnett, R.C., R.C. Martineau, and M. J. Lehto. “Preliminary Numerical Model of Radionuclide Transport in the Snake River Plain Aquifer Near the Idaho National Engineering Laboratory.” EGG–WM–8820. Idaho Falls, Idaho: EG&G Idaho, Inc. 1990.

Atkins, M. and F.P. Glasser. “Application of Portland Cement-Based Materials to Radioactive Waste Immobilization.” *Waste Management*. Vol.12. pp. 105–131. 1992.

Atkinson, A. and A. Nickerson. “Diffusion and Sorption of Cesium, Strontium, and Iodine in Water-saturated Cement.” *Nuclear Technology*. Vol. 81. p. 100–113. 1988.

Baker, K.E., L.C. Hull, T. McLing, L.V. Street, R.C. Roback, and C. Jones. “Fluid Transport Monitoring During Percolation Pond Infiltration and Recharge to a Complex, Deep, Heterogeneous Vadose Zone at the Idaho National Engineering and Environmental Laboratory.” Geological Society of American Annual Meeting. 2004.

Beasley, T. M., P.R. Dixon, and L.J. Mann. “^{99}Tc, ^{236}U, and ^{237}Np in the Snake River Plain Aquifer at the Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.” *Environmental Science and Technology*. Vol. 32. pp. 3,875–3,881. 1998.

Bradbury, M.H. and F.-A. Sarott. “Sorption Databases for the Cementitious Near-Field of a L/ILW Repository for Performance Assessment.” PSI Bericht 95-06. Würenlingen and Villigen, Switzerland: Paul Scherrer Institute. 1995.

Del Debbio, J. A. and T. R. Thomas. “Transport Properties of Radionuclides and Hazardous Chemical Species in Soils at the Idaho Chemical Processing Plant.” WINCO–1068. Idaho Falls, Idaho: Westinghouse Idaho Nuclear Company, Inc. 1989.

Doughty, C. “Numerical Model of Water Flow in a Fractured Basalt Vadose Zone; Box Canyon Site, Idaho.” *Water Resources Research*. Vol. 36, No. 12. pp. 3,521–3,534. 2000.

Faybishenko, B., C. Doughty, M. Steiger, J.C.S. Long, T.R. Wood, J.S. Jacobsen, J. Lore, and P.T. Zawislanski. “Conceptual Model of the Geometry and Physics of Water Flow in a Fractured Basalt Vadose Zone.” *Water Resources Research*. Vol. 36, No. 12. pp. 3,499–3,520. 2000.

Faybishenko, B., P.A. Witherspoon, C. Doughty, T. Geller, T. Wood, and R. Podgorney. “Multi-Scale Investigations of Liquid Flow in Fractured Basalt Vadose Zone, in Flow and Transport through Unsaturated Fractured Rock.” 2nd Edition. D. Evans, T.J. Nicholson, and T. Rasmussen, eds. *AGU Geophysical Monograph*. Vol. 42. 2001.

Fjeld, R. A., T. A. DeVol, R. W. Goff, M. D. Blevins, D. D. Brown, S. M. Ince, and A.W. Elzerman. “Characterization of the Mobilities of Selected Actinides and Fission/activation Products in Laboratory Columns Containing Subsurface Material from the Snake River Plain.” *Nuclear Technology*. Vol. 135. pp. 92–108. 2001.

Jones, C.L,. R.C. Roback, P.W. Reimus, H.J. Turin, E.M. Nichols, T.L. McLing, K.E. Baker, and L.C. Hull. "Flow and Transport in a Heterogeneous, Layered Vadose Zone at the Vadose Zone Research Park, Idaho National Engineering and Environmental Laboratory: Initial Results of Tracer Tests." Geological Society of America Annual Meeting. 2004.

Newman, M.E. and F.M. Dunnivant. "Results from the Large-Scale Aquifer Pumping and Infiltration Test: Transport through Fractured Media, Engineering Design File ER–WAG7–77." INEL–95/146. 1995.

NRC. SECY–03–0079, "NRC Review of Idaho National Engineering and Environmental Laboratory Draft Incidental Waste (Waste-Incidental-to-Reprocessing) Determination for Tank Farm Facility Closure." Washington, DC: NRC. 2003.

Porro, I., M.E. Newman, and F.M. Dunnivant. 2000. "Comparison of Batch and Column Methods for Determining Strontium Distribution Coefficients for Unsaturated Transport in Basalt." *Environmental Science and Technology*. Vol. 34. pp. 1,679–1,686. 2000.

Robertson, J.B. "Digital Modeling of Radioactive and Chemical Waste Transport in the Snake River Plain Aquifer at the National Reactor Testing Station, Idaho." U.S. Geological Survey Open-file Report IDO–22054. 1974.

Rodriguez, R.R., A.L. Shafer, J. McCarthy, P. Martian, D.E. Burns, D.E. Raunig, N.A. Burch, and R.L. VanHorn. "Comprehensive RI/FS for the Idaho Chemical Processing Plant OU 3-13 at the INEEL—Part A, RI/BRA Report (Final)." DOE/ID–10534, Binders 1–3. Idaho Falls, Idaho: DOE. 1997.

Scott, C., G. Heath, K. Baker, A. Schafer, and M. Shook. "Investigating Amendment Delivery Mechanisms for In Situ Stabilization of Strontium-90 in Porous and Fractured Media: Results of the Infiltration Test Created by the Flow of the Big Lost River." Geological Society of America Annual Meeting. 2005.

Sheppard, M.I. and D.H. Thibault. "Default Soil Solid/Liquid Partition Coefficients, K_ds, for Four Major Soil Types: A Compendium." *Health Physics*. Vol. 59. pp. 471–482. 1990.

Sorenson, K.S., A.H. Wylie, and T.R. Wood. "Test Area North Hydrogeologic Studies Test Plan: Operable Unit 1–07B." INEL–96/0105. Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory. 1996.

Wood, T. and B. Faybishenko. "Large-scale field investigations in fractured basalt in Idaho: Lessons learned, in Vadose Zone: Science and Technology Solutions." B.B. Looney and R.W. Falta, eds. Vol. 1. pp. 396–405. 2001.