U. S. NUCLEAR REGULATORY COMMISSION TECHNICAL EVALUATION REPORT FOR THE U.S. DEPARTMENT OF ENERGY IDAHO NATIONAL LABORATORY SITE DRAFT SECTION 3116 WASTE DETERMINATION FOR IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER TANK FARM FACILITY

October 2006

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LIST OF ABBREVIATIONS, ACRONYMS, AND NOMENCLATURE

ADAMS	Agencywide Document Access and Management System
ALARA	As Low As Is Reasonably Achievable
Am	americium
Ba	barium
Bq	becquerel
BLR	Big Lost River
C	carbon
°C	degree Celsius
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFA	Central Facilities Area
CFR	Code of Federal Regulations
cfs	cubic feet per second
Ci	curie
Cm	curium
cm	centimeter
cm^3	cubic centimeters
Co	cobalt
Cs	cesium
d	day
DCF	dose conversion factor
DEQ	Department of Environmental Quality
DOE	U.S. Department of Energy
DOE Idaho	U.S. Department of Energy, Idaho Operations Office
DQA	data quality assessment
DQO	data quality objective
DUST-MS	Disposal Unit Source Term Multiple Species
DSA	documented safety analysis
EDF	engineering design file
EPA	U.S. Environmental Protection Agency
Eu	europium
°F	degrees Fahrenheit
FORTRAN	formula translation/translator (high-level programming language)
FR	Federal Register
ft	foot
ft^3	cubic feet
g	grams
gal	gallon
GWSCREEN	A Semi-Analytical Model for Assessment of the Groundwater Pathway from Surface or Buried Contamination
H-3	tritium
ha	hectare
HCM	hydrogeological conceptual model
HLW	high-level radioactive waste
hr	hour
HRRs	highly radioactive radionuclides
HWMA	Hazardous Waste Management Act
I	iodine

LIST OF ABBREVIATIONS, ACRONYMS, AND NOMENCLATURE (CONTINUED)

ICRP	International Commission on Radiological Protection
INEEL	Idaho National Engineering and Environmental Laboratory
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
in	inches
K_d	distribution coefficient
K_h	hydraulic conductivity
km	kilometers
L	liters
LNT	linear no-threshold
LLW	low-level radioactive waste
m	meters
m^3	cubic meters
MBq	mega becquerel
MCi	million curies
MCL	maximum contaminant level
MCP	management control procedure
mi	miles
mL/g	milliliters/gram
MOU	Memorandum of Understanding
mph	miles per hour
mrem	millirem
mSv	millisievert
NAS	National Academy of Sciences
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
Nb	niobium
nCi	nano Curie
Ni	nickel
Np	neptunium
NRC	U.S. Nuclear Regulatory Commission
ORIGEN2	Isotope Generation and Depletion Code
oz	ounce
PA	performance assessment
pCi/L	picocuries/liter
pH	measure of acidity (minus the log of the hydrogen ion concentration)
PMF	probable maximum flood
PMP	probable maximum precipitation
PNNL	Pacific Northwest National Laboratory
PORFLOW	Code Used to Model Multiphase Fluid Flow, Heat and Mass Transport in Variably Saturated Porous and Fractured Media
Pu	plutonium
QA	quality assurance
RAI	request for additional information
RCRA	Resource Conservation and Recovery Act
rem	unit of dose equivalent
RI/BRA	remedial investigation/baseline risk assessment

LIST OF ABBREVIATIONS, ACRONYMS, AND NOMENCLATURE (CONTINUED)

RPP	Radiological Protection Plan
s	second
SAP	sampling and analysis plan
Sb	antimony
SBW	sodium-bearing waste
SNF	spent nuclear fuel
Sr	strontium
SRM	Staff Requirements Memorandum
SRP	standard review plan
SRPA	Snake River Plain Aquifer
Sv	Sievert
Tc	technetium
TEDE	total effective dose equivalent
TER	Technical Evaluation Report
TFA	Tank Focus Area
TFF	Tank Farm Facility
TRU	transuranic
U	uranium
USGS	U.S. Geological Survey
WIR	waste incidental to reprocessing
Y	yittrium
yr	year

EXECUTIVE SUMMARY

In September 2005, the U.S. Department of Energy (DOE) submitted the "Draft Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center (INTEC) Tank Farm Facility (TFF)" for review by the U.S. Nuclear Regulatory Commission (NRC), as required by the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA). Section 3116 of the NDAA requires DOE to consult with the NRC when determining that certain wastes associated with spent fuel reprocessing are not high-level wastes. The draft waste determination addresses TFF waste that DOE proposes to grout and dispose of in place. This Technical Evaluation Report (TER) presents information on DOE's disposal strategy, the applicable review criteria, and NRC staff's review approach, as well as NRC staff's analysis and conclusions with respect to whether there is reasonable assurance that DOE's proposed approach can meet the applicable NDAA criteria. NRC is not providing regulatory approval of DOE's waste determination activities. DOE is responsible for determining whether the waste streams addressed in the draft waste determination are not high-level waste (HLW).

Based on the information provided by DOE, NRC staff has concluded in this TER that there is reasonable assurance that the applicable criteria of the NDAA can be met for residual waste associated with the TFF. The NDAA requires NRC, in coordination with the State of Idaho, to monitor DOE disposal actions to assess DOE compliance with the performance objectives in Title 10, Code of Federal Regulations, Part 61 (10 CFR Part 61), Subpart C. During its review of DOE's draft waste determination, NRC identified key monitoring areas that are important for DOE to meet the performance objectives (see Section 4.4).

There are 15 HLW tanks at the INL site: four are 100-m^3 [30,000-gal] tanks and the remainder (eleven tanks) are larger 1,000-m^3 [300,000-gal] tanks. Seven of the large tanks and all of the small tanks have been cleaned. As of April 30, 2005, the large cleaned tanks have several thousand gallons of liquid waste in each tank, while three out of four of the large tanks that have not been cleaned are nearly filled to capacity. One spare tank that has not been cleaned has a few hundred gallons of liquid waste remaining. TFF closure includes cleaning and stabilization activities for tank system components (including tanks, vaults, piping, structures, and ancillary equipment). DOE proposed to reduce the remaining residual waste volume in each tank to a depth of approximately 3 cm [1 in] prior to adding reducing grout to the tanks to stabilize the waste. The total estimated activity expected in the tank system at closure is 9.6×10^{14} Bq [2.6×10^4 Ci] (decayed to the year of expected closure, 2012). The tank waste consists of a highly acidic mixture of residual solid and liquid waste residuals that are easily dispersible. The first-cycle extraction waste and most of the second- and third-cycle extraction waste from HLW reprocessing that was originally stored in the tanks has been calcined[1] in the New Waste Calcining Facility at the INL site. Neither the calcined waste nor the New Waste Calcining Facility is part of the INL TFF waste determination. The remaining sodium-bearing waste (SBW) stored in the TFF contains primarily second- and third-cycle extraction waste and decontamination fluids.

U.S. Department of Energy, Idaho Operations Office (DOE Idaho) has used a cleaning system consisting of a washball and directional nozzle and a modified steam jet pumping system to

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[1] Waste calcining is a process whereby liquid radioactive wastes are turned into a smaller volume of granular solid so less storage space is required.

remove residual liquid waste and mobilize residual solid heel[2] to the jet pumps for bulk removal. Dissolution of more soluble radionuclides in the solid residual waste into the cleaning water that is pumped from the tanks is also a removal mechanism. Tank cleaning activities completed thus far have demonstrated the effectiveness of the cleaning system. The estimated remaining activity in the tanks is approximately 10 percent of the activity estimated by DOE prior to significant cleaning activities. DOE Idaho estimates that over 99.9 percent of the total inventory stored in the TFF during its operational history is expected to be removed prior to closure.

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

The second criterion of the NDAA is that the waste has had highly radioactive radionuclides (HRRs) removed to the maximum extent practical. To assess conformance with Criterion Two, the NRC staff assessed DOE Idaho's estimated waste inventory, identification of HRRs, selection of treatment technology, and demonstration of removal to the maximum extent practical including the costs and benefits of additional radionuclide removal.

NRC staff's conclusions regarding Criterion Two are based on the following assumption:

- Inventory estimates for the large tanks that have not been cleaned (WM–187 through WM–190) are not significantly underpredicted (i.e., similar or better waste retrieval will be achieved than is currently assumed by DOE Idaho).

NRC staff's conclusions with respect to Criterion Two are the following:

There is reasonable assurance that Criterion Two of the NDAA can be met because:

- The estimated inventory developed for the tanks and sand pad in the performance assessment and validated through sampling for the tanks in the waste determination is reasonable for the purpose of evaluating compliance with NDAA criteria.

- HRRs have been or will be removed to the maximum extent practical based on an evaluation of DOE's selection of HRRs; DOE's selection, implementation and effectiveness demonstration for its preferred cleaning technology; and the NRC staff evaluation of the costs and benefits of additional removal.

[2]A tank heel refers to the remaining waste in each tank after lowering the level to the greatest extent possible using existing waste transfer equipment such as steam jet pumps.

The third criterion of the NDAA is that waste will be disposed of in compliance with 10 CFR Part 61, Subpart C, performance objectives. If the waste is greater than Class C based on concentration limits provided in 10 CFR 61.55, the NDAA requires DOE to further consult with the NRC regarding its disposal plans. Subpart C to 10 CFR Part 61 sets requirements for protection of the public, the inadvertent intruder, and individuals during operations, and also provides for site stability. To assess conformance with Criterion Three, NRC staff evaluated DOE Idaho's (i) waste classification for tank system components; (ii) performance assessment (including infiltration, near-field release, far-field transport, dose methodology, and exposure assessment); (iii) inadvertent intruder analysis; (iv) radiation protection program for individuals during operations; and (v) stability of the disposal facility after closure. The NDAA also requires disposal of waste pursuant to state-approved closure plans or state-issued permits. DOE Idaho is seeking clean-closure status with the Idaho Department of Environmental Quality (DEQ) for the tanks at INTEC under the Hazardous Waste Management Act (HWMA) and Resource Conservation and Recovery Act (RCRA) which would allow DOE Idaho to close the TFF without a state permit.

The NRC staff's conclusions regarding Criterion Three are based on the following assumptions:

- Active institutional controls will be maintained for 100 years.

- The model limitations or uncertainties NRC staff identified in DOE Idaho's hydrogeological conceptual model (HCM) and hydrogeologic model construction and implementation will not significantly alter the conclusions in this TER.

- Inventory estimates for the large tanks that have not been cleaned (WM–187 through WM–190) are not significantly underpredicted (i.e., similar or better waste retrieval will be achieved than is currently assumed by DOE Idaho).

The NRC staff's conclusions with respect to Criterion Three are the following:

There is reasonable assurance that DOE Idaho can meet Criterion Three of the NDAA because:

- Based on information provided by DOE Idaho, NRC staff expects the maximum public dose from all pathways to be below the 0.25-mSv/yr [25-mrem/yr] dose limit. In addition, DOE's waste determination states that reasonable effort will be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonable achievable. Therefore NRC staff concludes that there is reasonable assurance that DOE Idaho can meet 10 CFR 61.41 requirements.

- Based on analysis provided by DOE Idaho, NRC staff concludes that there is reasonable assurance that DOE Idaho can meet 10 CFR 61.42 requirements for protection of individuals from inadvertent intrusion.

- Workers are protected by DOE regulations that are comparable to 10 CFR Part 20. DOE Idaho controls are also in place to protect members of the public during operations. Therefore, NRC staff concludes that there is reasonable assurance that DOE Idaho can meet 10 CFR 61.43 requirements for protection of individuals during operations.

- DOE Idaho plans to fill the tanks, vaults, and ancillary equipment with grout which will provide structural stability and limit waste dispersal. Therefore, the NRC staff concludes that there is reasonable assurance that DOE Idaho can meet 10 CFR 61.44 stability requirements.

For a broader and more detailed discussion of DOE Idaho's approach and NRC staff's analysis and conclusions, please see the appropriate sections of the TER. All of the conclusions reached by the NRC staff are based on DOE Idaho's Draft Section 3116 Waste Determination dated September 7, 2005; DOE Idaho's responses to NRC's request for additional information; supporting references; and information provided during meetings between DOE and NRC. If in the future DOE determines it is necessary to revise its assumptions, analysis, design, or waste management approach and those changes are important to meeting the criteria of the NDAA, DOE should consult once again with NRC. Note that NRC is providing consultation to DOE as required by the NDAA, and NRC is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW. This NRC staff assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental waste determinations at INL or other DOE sites.

1 INTRODUCTION

The Idaho Nuclear Technology and Engineering Center (INTEC) Tank Farm Facility (TFF) is located on the Idaho National Laboratory (INL) Site. The INL is an approximately 2,305 km^2 [890 mi^2] reservation owned by the United States government and located in southeastern Idaho (see Figure 1). The INTEC facility is located approximately 29 km [18 mi] from the closest eastern boundary, approximately 23 km [14 mi] from the closest western boundary, approximately 16 km [10 mi] from the closest southern boundary, and approximately 29 km [18 mi] from the closest northern boundary. The TFF, located within the northern portion of INTEC, comprises eleven 1,000-m^3 [300,000-gal] below-grade stainless steel tanks in unlined concrete vaults of various construction, four inactive 100-m^3 [30,000-gal] stainless steel tanks, interconnecting waste transfer lines, and associated support instrumentation and valves. Structures located above ground level in the TFF include the TFF Control House, the Computer Interface Building, and the tank and vault sump riser covers. A perimeter fence encloses the TFF (see Figures 2 and 3).

Established in 1953, the Idaho Chemical Processing Plant, now INTEC, was chartered to recover fissile uranium by reprocessing spent nuclear fuel (SNF). The spent fuel was dissolved, producing an acidic aqueous solution that was processed through a first-cycle extraction system to separate uranium from the bulk of the fission products (or first-cycle extraction waste). The separated uranium was processed through a second- and third-cycle extraction system to remove carry-over radioactive material, which included plutonium and transuranic radionuclides. In 1992, the DOE officially discontinued reprocessing SNF at INTEC, and the first cycle extraction process wastes stored in the TFF were removed and solidified by calcination (a thermal process whereby liquids are converted to solid oxides) by February 1998. Historically, the TFF tanks were used to store various INTEC wastes, including those from SNF reprocessing (first-, second-, and third-cycle reprocessing wastes), decontamination waste, laboratory waste, and contaminated liquids from other INTEC operations. In general, because of significantly higher radioactivity levels, first-cycle reprocessing wastes were segregated from the other types of liquid waste. These other tank wastes, referred to as sodium-bearing waste (SBW) because of their high sodium levels, were made up of lower activity wastes other than first-cycle reprocessing wastes, and had a significantly different chemical composition than first-cycle reprocessing wastes.

Placed into service between 1953 and 1966, the eleven 1,000-m^3 [300,000-gal] tanks (WM–180 through WM–190) are approximately 15.2 m [50 ft] in diameter and 6.4–7.0 m [21–23 ft] in height. Nine of the eleven 1,000-m^3 [300,000-gal] tanks are constructed of Type 304L stainless steel and the other two tanks (WM–180 and WM–181) use Type 347 stainless steel. Each tank has four or five 30 cm [12 in] diameter risers to provide access to the tank. Tanks WM–184 through WM–190 also have one or two 46 cm [18 in] diameter risers. All eleven 1,000-m^3 [300,000-gal] tanks are housed in reinforced concrete vaults with the bottom of the vault approximately 13.7 m [45 ft] below grade. The tanks rest on a 15 cm [6 in] layer of sand distributed over the bottom of the vaults. The vaults have different shapes: two octagon, five pillar and panel, and four square vaults. Placed into service in 1956, the four 100-m^3 [30,000-gal] stainless steel below-grade storage tanks (WM–103 through WM–106) sit on reinforced concrete pads and were removed from service in 1983. The tanks are horizontal cylinders approximately 3.5 m [11.5 ft] in diameter and 11.6 m [38 ft] in length. The 100-m^3 [30,000-gal] tanks do not have vaults or sand pads.

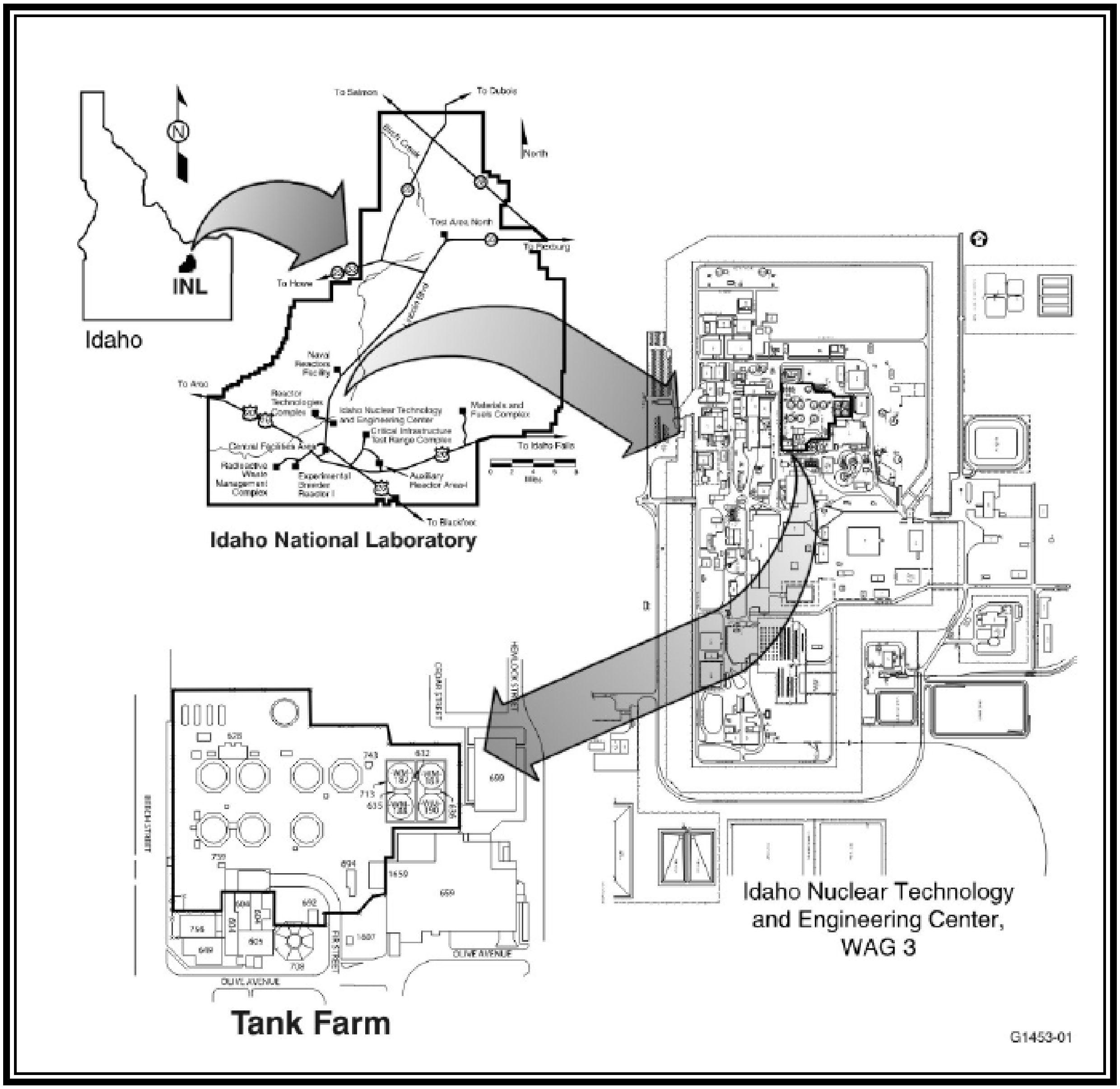

Figure 1. Idaho National Laboratory (From Figure A-1-1, DOE Idaho 2006e)

DOE intends to close the TFF in phases to support continued INTEC operations. The closure process comprises tank system cleaning and stabilization activities. The TFF equipment and structures that are potentially contaminated with reprocessing wastes as a result of past INTEC reprocessing operations include the stainless steel tanks, concrete vaults, sand pads, piping, encasements, valve boxes, and instrumentation lines. These structures, systems, and components will be isolated and grouted as a part of the INTEC TFF final closure.

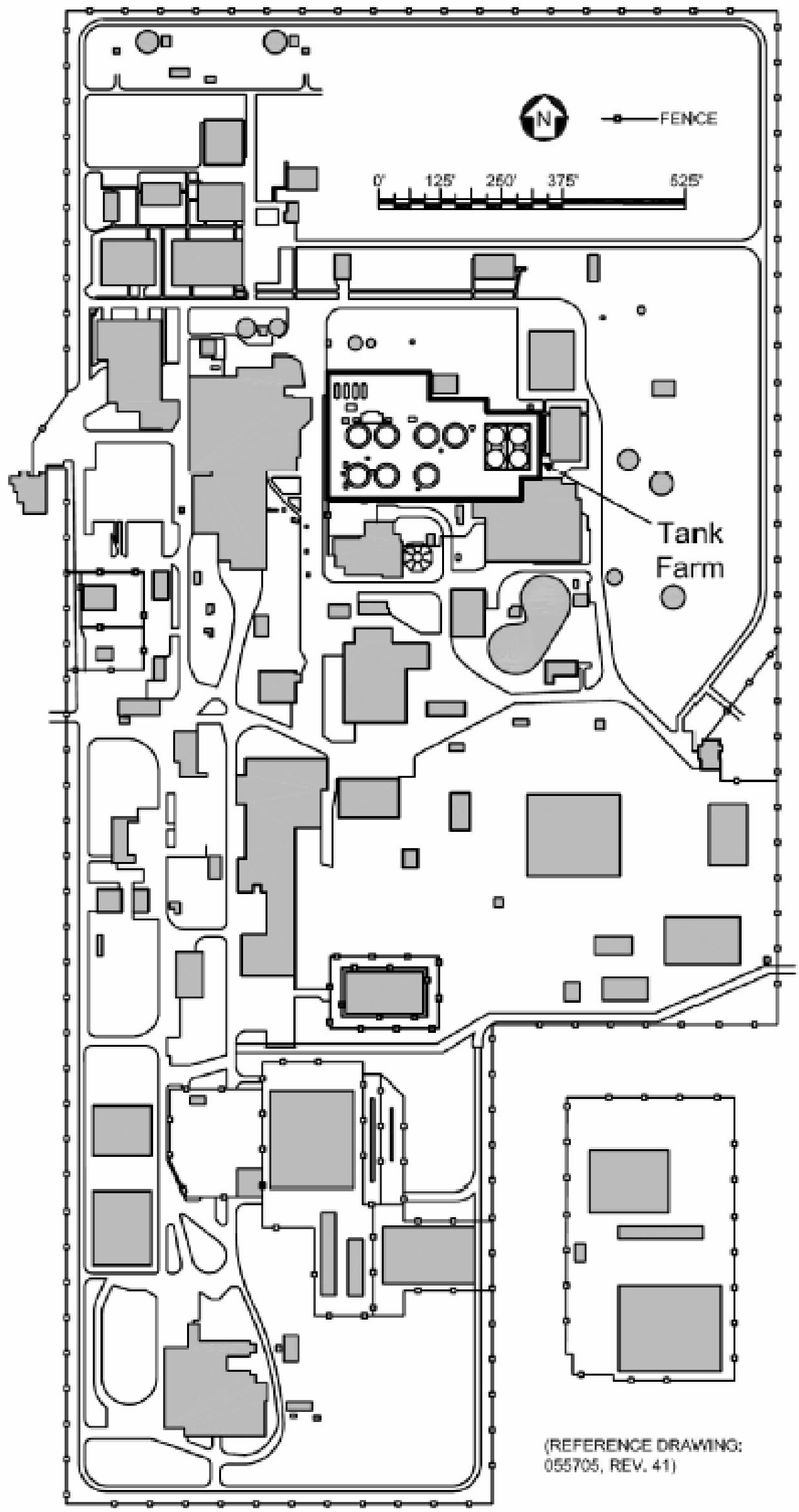

Figure 2. Tank Farm Facility at INTEC (From Figure 2, DOE Idaho, 2005)

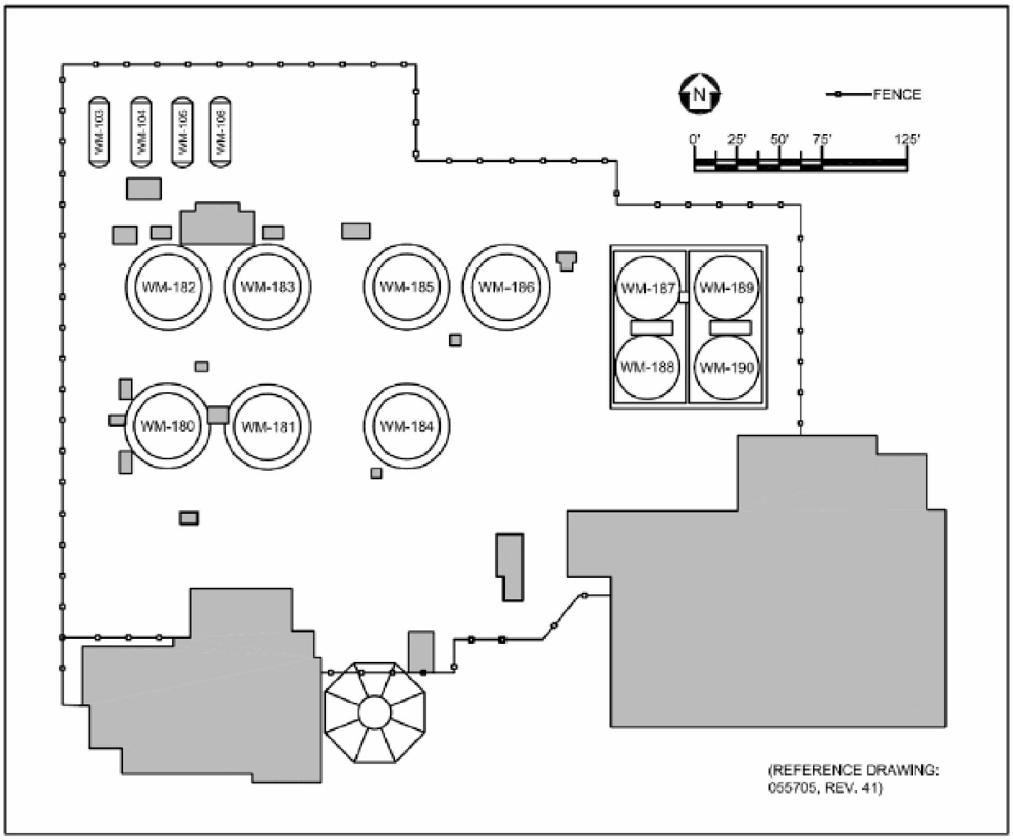

Figure 3. Plan View of the Tank Farm Facility (From Figure 3, DOE Idaho, 2005)

Tank cleaning began in 2002, and the final TFF closure is planned for 2012. To date, seven of the 1,000-m^3 [300,000-gal] tanks, the four 100-m^3 [30,000-gal] tanks, and associated ancillary equipment have been cleaned. The remaining approximately 3,000 m^3 (900,000 gal) of SBW are stored in three 1,000-m^3 [300,000-gal] tanks. Additionally, one 1000-m^3 [300,000-gal] tank is maintained as a spare (see Figure 4) and has only a few hundred gallons of waste.
U.S. Department of Energy, Idaho Operations Office (DOE Idaho) assumes that the same degree of radionuclide removal will be achieved in the remaining four tanks as that achieved in the seven tanks that have been cleaned. After cleaning activities are completed for all of the tanks and ancillary equipment in the TFF, DOE Idaho plans to stabilize the TFF by filling the tank system with grout for final closure. DOE Idaho's approach for cleaning and stabilizing the TFF is evaluated by the U.S. Nuclear Regulatory Commission (NRC) staff in this Technical Evaluation Report (TER).

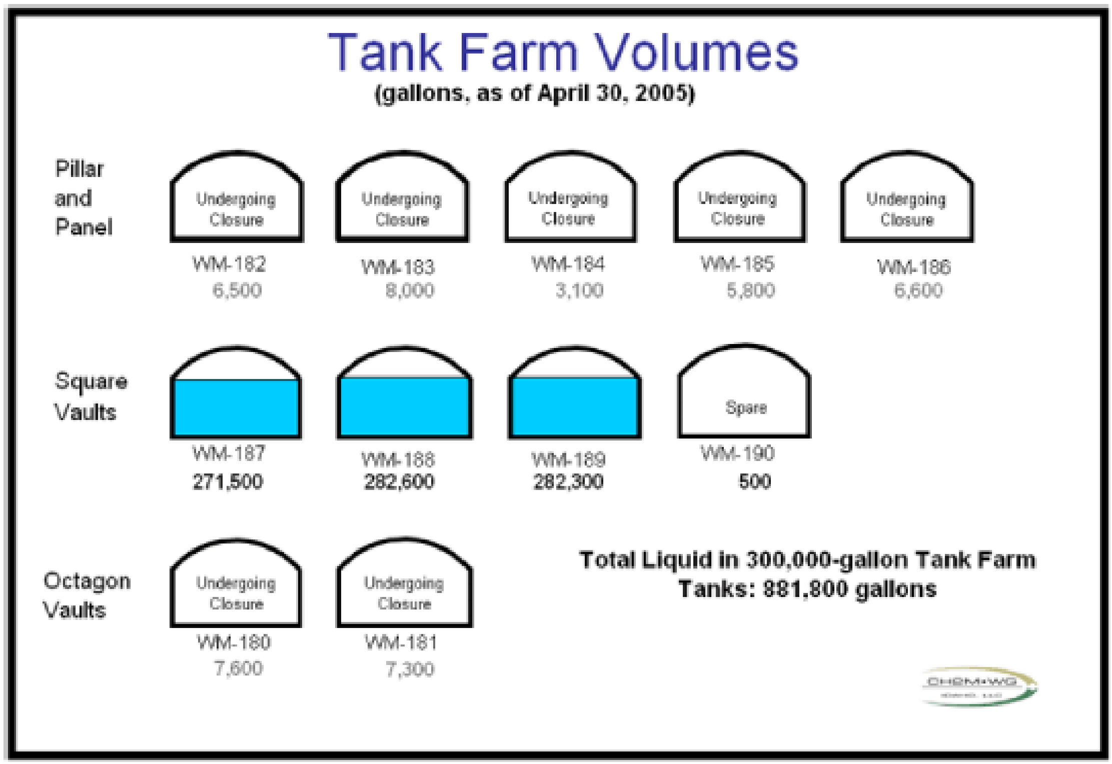

Figure 4. Waste Volumes Remaining in Large Tanks (From Table 9, DOE Idaho, 2005)

1.1 Facility and Site Description

1.1.1 Facility Description

The TFF is located within the boundary of the INTEC on the INL site. INTEC is geographically located in Butte County, Idaho, approximately 47 km [29 mi] west of Idaho Falls, Idaho. INL and INTEC are located in a broad, relatively flat plain in the Pioneer Basin—a closed drainage basin. INTEC lies within a perimeter fence enclosing approximately 80 hectare (ha) [200 acres]. The TFF is approximately 460 m [1,500 ft] from the Big Lost River (BLR) channel. The BLR is an intermittent flowing stream that sinks into the permeable vadose zone and the Snake River Plain Aquifer (SRPA) below.

The primary missions of the facilities at INTEC are to safely store SNF, prepare SNF for permanent storage in an offsite repository, develop technologies for safe treatment of high-level and liquid radioactive waste from reprocessing SNF, and remediate any past environmental release of radioactive materials (Foster Wheeler Environmental Corporation, 2003a).

INL is designated as an exclusion area for nuclear reactors and associated facilities and thus is isolated to ensure maximum public safety. There are no permanent residents within an 18 km [11 mi] radius of INTEC.

The following site background information provides context for the evaluation sections that follow, particularly Section 4 on performance objectives.

1.1.2 Site Description

1.1.2.1 Land and Water Use

The "Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory" (PA; DOE Idaho, 2003a) states that land use at the INL is currently government-controlled industrial use. Because INL is controlled, the public does not have unrestricted access to any of the facilities on the INL site. These controls are assumed to be in place for a minimum of 100 years. The PA (DOE Idaho, 2003a) assumed that the institutional control period would begin at the end of closure for the TFF, which was originally planned for 2016 (when the DOE PA was written) but has since changed to 2012.

Categories of land use at INL include facility operations, grazing, general open space, and infrastructure (e.g., roads). Facility operations include industrial and support operations associated with energy research and waste management activities. Much of INL is open space that is not designated for specific uses. Some of this space serves as a buffer zone between the INL facilities and other land uses. Between 121,000 and 142,000 ha [300,000 and 350,000 acres] of INL land are used for cattle and sheep grazing. Grazing is not allowed within 3 km [2 mi] of any INL nuclear facility, and to avoid the possibility of milk contamination by long-lived radionuclides, dairy cattle are not permitted. Approximately 2 percent {4,600 ha [11,400 acres]} of INL is used for facilities and operations. Approximately 6 percent {13,870 ha [34,260 acres]} of INL is devoted to public roads and utility rights-of-way that cross INL. Because INL is remote from most developed areas, the INL lands and adjacent areas are not likely to experience extensive residential and commercial development. However, a DOE Idaho Operations Office (DOE Idaho) study showed recreational and agricultural uses would increase in the surrounding area because of increased demand and the conversion of range land to crop land (DOE Idaho, 1993).

DOE Idaho (1995a) describes the surface and subsurface water use in the affected environment at INL. INL does not withdraw or use surface water for site operations, nor does it discharge effluents to natural surface water. The three surface-water bodies at or near the site (Big and Little Lost Rivers and Birch Creek), however, have the following designated uses: agricultural water supply, cold-water biota, salmonid spawning, and primary and secondary contact recreation. In addition, waters in the Big Lost River (BLR) and Birch Creek have been designated for domestic water supply and as special resource waters. The Snake River Plain Aquifer (SRPA) is the only source of water used at INL, and INTEC wells withdraw water from this aquifer. The water withdrawn from each well is used for potable water, ground maintenance, and necessary INTEC operations. On the regional scale, water from the aquifer is used for agriculture; food processing; aquaculture; and domestic, rural, public, and livestock water supplies (DOE Idaho, 2003a). In total, nearly 18 trillion L [4.7 trillion gal] are drawn from the aquifer annually, with the majority used for agriculture (DOE Idaho, 1998a). The PA documentation (DOE Idaho, 2003a) states that the Eastern SRPA has been designated by U.S. Environmental Protection Agency (EPA) as a sole source aquifer (58 Federal Register 138, 1991). After sole source designation, any federal financial assistance projects are subject to EPA approval to ensure that they do not contaminate the aquifer and create a significant hazard to public health.

1.1.2.2 Terrestrial and Aquatic Biota

As discussed in the PA (DOE Idaho, 2003a), the INL site was dedicated as one of five DOE National Environmental Research Parks in 1975. The INL site is used to study ecological systems, the changing environment over time, and the impact of human activities on the environment. Research on the ecology at INL has been performed with the DOE Radiological and Environmental Sciences Laboratory. The physical attributes and types of flora and fauna present on the site are typical of cold, high altitude, sagebrush ecosystems found in many parts of the western United States (DOE Idaho, 2003a).

Vegetation at INL is limited by soil type, meager rainfall, and extended drought periods. Native plants consist mainly of sagebrush (*Artemisia tridentata*) and a variety of grasses. Lanceleaf rabbitbrush (*Crysothamnus viscidiflorus*) is common. The INTEC area is kept free from vegetation such that there is limited fuel for range fires (Foster Wheeler Environmental Corporation, 2003a). The DOE PA lists the findings of several studies where the roots of big sagebrush extended to a depth of 225 cm [88.7 in], green rabbitbrush to a depth of 190 cm [74.9 in], and Great Basin wildrye up to 200 cm [78.8 in] deep at the Subsurface Disposal Area. Maximum lateral spread of the roots of both big sagebrush and Great Basin wildrye was 90.2 cm [35.5 in] and occurred at a depth of 40.1 cm [15.8 in]. In addition, studies indicate root penetration of up to 160 cm [62.4 in] for sodar and crested wheatgrass at the INL (DOE Idaho, 2003a).

Fauna commonly occurring at the INL site includes mammals (e.g., chipmunks, ground squirrels, several species of mice, kangaroo rats, cottontail rabbits, bats, jackrabbits, and coyotes); game animals (e.g., sage grouse, mourning dove, elk, pronghorn, and mule deer); fish (observed in the BLR on the INL, e.g., rainbow trout, mountain whitefish, eastern brook trout, Dolly Varden char, Kokanee salmon, and shorthead sculpin); amphibians and reptiles (e.g., spadefoot toad, sagebrush lizard, short-horned lizard) and various snakes; and birds (e.g., Brewer’s sparrow, sage thrasher, sage sparrow, horned lark, mourning dove, western meadowlark, blackbilled magpie, and robin). Several threatened and endangered species occur on the INL site. Results of the studies listed in the DOE PA indicate burrows no deeper than 140 cm [55 in] at the INL site (DOE Idaho, 2003a).

1.1.2.3 Local Meteorology and Climatology

The highest and lowest daily maximum temperatures at the Central Facilities Area [located approximately 5 km [3 mi] south of INTEC] range from 38.3 °C [101 °F] in July to −44 °C [−47 °F] in December. The average annual temperature at the INL exhibits a gradual 7-month increase, beginning with the first week in January and continuing through the third week in July. The temperature then decreases over five months until the minimum average temperature is reached again (DOE Idaho, 2003a). From April through October, the average monthly temperature varies from 41 to 68 ˚F [5 to 20 ˚C].

The average annual precipitation at the Central Facilities Area is 22.15 cm [8.72 in]. The maximum recorded annual precipitation was 36.6 cm [14.4 in] in 1963, and the minimum recorded annual precipitation was 11 cm [4.5 in] in 1966. A monthly precipitation peak of approximately 3.0 cm [1.2 in] is associated with thunderstorms in May and June each year (DOE Idaho, 1995b). Other months generally receive one-half or less of this amount. Snowfall ranges from 17 to 200 cm/yr [6.7 to 78.7 in/yr] with an annual average of 71 cm [28 in]. The maximum average monthly snowfall is 16 cm [6.4 in], occurring in December. Between one-quarter and one-third of the average annual precipitation is contributed in the form of snow (Sehlke and Bickford, 1993).

The prevailing wind direction at INTEC, and most locations at INL, is southwesterly. The average monthly wind speed varies from 4.9 km/h [3.1 mph] in December to 15 km/h [9.3 mph] in April and May. The highest hourly average speed was recorded from the west-southwest at 82 km/h [51 mph]. Calm conditions prevail 11 percent of the time (DOE Idaho, 2003a).

Topographic maps indicate there is approximately 1 m [3 ft] of relief across the TFF. The expected meteorologic environment and minimal topographic relief suggest that the site will not experience significant erosion.

1.1.2.4 Hydrology and Hydrogeology

The BLR is the only stream with potential for having an impact on the TFF. The BLR enters INL in the southern portion of its western boundary, and in the wettest years, flows east and north in an arc to the foot of the Lemhi Mountain Range where it ends in the BLR playas or sinks. Episodic pulses of water from spring snow-melt may drive water and solutes deeper into the subsurface over a matter of just days to weeks. DOE Idaho reviewed historical information and found that there has not been a case of inundation from storms or runoff to cause flooding of the INTEC site since 1952 (DOE Idaho, 2003a). There is evidence of prehistoric flooding in the geologic sediments at the site.

The estimated 100-year peak flow of the BLR immediately upstream of the INL Diversion Dam is 106 m^3/s [3,750 cfs] with upper and lower 95-percent confidence limits of 177 m^3/s [6,250 cfs] and 37 m^3/s [1,300 cfs] (Hortness and Rousseau, 2003). The estimated 100-year peak flow of the BLR at INTEC is 82.4 m^3/s [2910 cfs] (Parker, 2006). DOE Idaho commissioned additional studies to refine the 100-year flood plain and delineate the 500-year flood plain, including a two-dimensional model analysis and a paleohydrologic and geomorphic assessment of the flood risk along the BLR (Ostenaa, et al., 1999), consistent with requirements contained in the DOE standards for a comprehensive flood hazard assessment. The probable maximum flood (PMF) due to Mackay Dam failure bounds this 100-year flood.

INTEC is underlain by the alluvial veneer of Pleistocene-to-Holocene BLR floodplain material and a sequence of Quaternary volcanic rocks and sedimentary interbeds (Whitehead, 1992). The TFF is embedded in 13 m [43 ft] of alluvial silt, sand, and gravel that lie above an alternating sequence of basalt lava flows and interbedded sediments (see Figure 5, DOE Idaho, 2006g). The volcanic rocks in the upper portion of the subsurface that are relevant to this waste determination consist of basalt flows, volcanic vent deposits, and dikes (Anderson, et al., 1999). Basalts and sediments generally range in age from approximately 200,000 to 640,000 years before present.

Distinct basalt flows generally range from 3 to 20 m [10 to 66 ft] thick and are internally interbedded with discontinuous scoria and thin layers of sediment. Sedimentary interbeds accumulated above basalt flows for hundreds to hundreds of thousands of years during periods of volcanic quiescence; interbeds are, therefore, thickest between basalt-flow groups because each flow group is associated with a specific eruptive event. Sedimentary interbeds were likely deposited in eolian or fluvial environments and may be as thick as 20 m [66 ft]. Sedimentary interbeds are known to consist of silt, sand, gravel, small clay lenses, scoria, and basalt rubble. Most wells at and near INTEC are completed in the vadose zone and upper SRPA basalts and sediments (e.g., Figure 2-12, DOE Idaho, 2003a). Stratigraphic sections derived from well characterization activities indicate that basalt flows and sedimentary interbeds comprise horizontal to slightly inclined layers. More than 30 geologic units comprise the vadose zone and the upper portion of the SRPA, including 19 flow groups, 11 sedimentary interbeds, and surficial alluvium (see for instance, Figure 2-12, DOE Idaho, 2003a).

Several perched water zones underlie the INTEC facility (see Figure 5); these zones are commonly divided into upper and lower perched water zones. The upper perched water zones

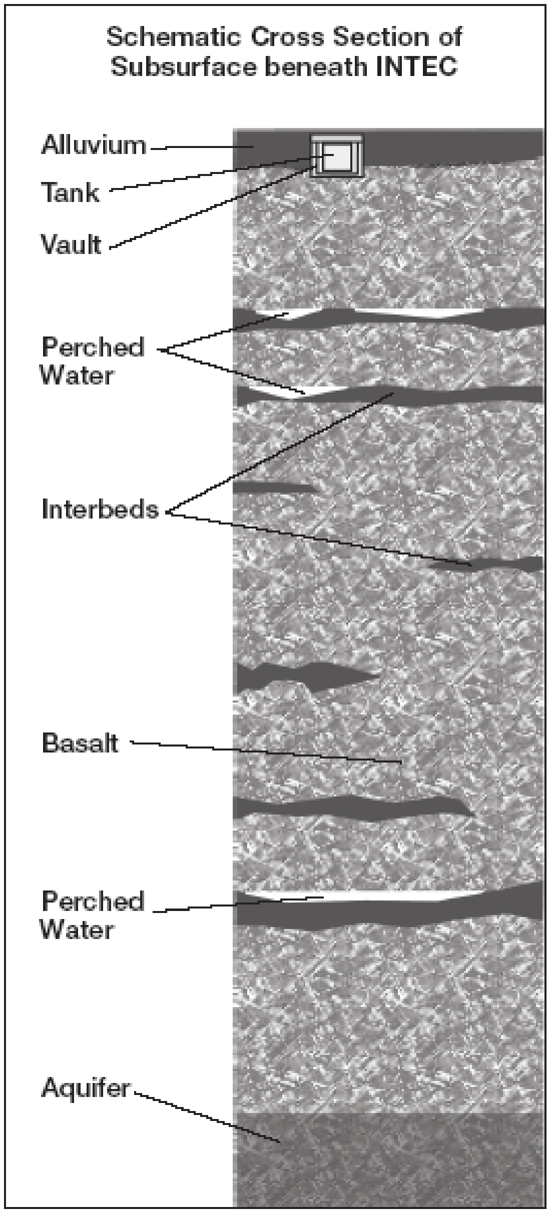

Figure 5. Schematic of the Subsurface at the TFF (Not to the Scale, Taken From DOE Idaho, 2006g)

are defined as perched water that occurs between the ground surface and 58 m [190 ft] below the ground surface (see Figure 2-21, DOE Idaho, 2003a). The lower perched water zones are defined as perched water that occurs at elevations between 98 and 130 m [320 and 420 ft] below the ground surface (see Figure 2-22, DOE Idaho, 2003a). Occurrences of perched water at this site correspond to flow barriers created by low permeability portions of sedimentary interbeds, and the perching of water in the vadose zone is brought about primarily by operational processes and landscape watering, and secondarily by ephemeral BLR seepage that releases water into the subsurface at INTEC.

The SRPA is one of the largest and most productive aquifers in the United States, receiving natural recharge at and near INTEC from precipitation and BLR underflow (Bennett, 1990). Underflow is water flow that occurs in the permeable fluvial deposits directly beneath the BLR channel whether or not there is sufficient water for surface flow (Smith, 2004); underflow cannot be measured by a stream gauge. Anthropogenic sources of recharge to the SRPA on INL property include process waste water disposed of in percolation ponds and a disposal well, piping leaks, and landscape irrigation water. Some of these sources of water are being addressed by an interim action for the TFF (see “existing contamination” discussion within this section below). Recharge from the BLR has caused water levels in some wells to rise as much as 2 m [6 ft] in a few months after high flows in the river (Barraclough, et al., 1982). However, recent information suggests that BLR flow is not the most significant source of water for the upper shallow perched water at the TFF, because there was minimal well response following flow of the BLR in 2005 (DOE Idaho, 2006e). Changes in perched water levels appear to be caused by rain and snowmelt infiltration rather than BLR flow (DOE Idaho, 2006e). The direction of groundwater flow below the BLR is directly affected by recharge from the BLR. The BLR channel trends approximately 31 degrees north of east near the TFF.

Volcanic vents are the source of Eastern Snake River Plain basalt flow groups. Vent and near-vent deposits are highly permeable hydrologic features (Anderson, et al., 1999; Smith, 2004). These features are concentrated near several northwest-trending linear volcanic rift zones and within the northeast-trending Axial Volcanic Zone, which is located along the central axis of the Eastern Snake River Plain (Figure 2-15, DOE Idaho, 2003a). Volcanic dikes and thick, tube-fed pahoehoe flows and flows that ponded inside vent craters and topographic depressions are impermeable hydrologic features (Anderson, et al., 1999; Smith, 2004).

Volcanic dikes in the Snake River Plain are vertical features that strike northwest to southeast; dikes crosscut lava flows in both the aquifer and vadose zone (Smith, 2004). In addition to vents and dikes, the hydrostratigraphy of the unconfined SRPA is comprised of massive basalt flows and thinner overlying basalt flows (see Figure 2-12, DOE Idaho, 2003a). The hydraulic conductivity of the SRPA at INL was estimated from 114 single-well aquifer tests to range six orders of magnitude from 3.0×10^{-3} to 9.8×10^{3} m/d [1.0×10^{-2} to 3.2×10^{4} ft/d] (Anderson, et al., 1999), with low values being evidence for the presence of dikes and high values being representative of near-vent deposits. Groundwater flow velocities near INTEC have been estimated to be 1.5 m/d [5 ft/d]. Groundwater flow occurs through fractures (joints) in the basalt and along rubble zones at flow contacts (bedding planes). Permeabilities decline with depth in the aquifer. Current best estimate effective porosity of the SRPA is 3 percent, based on calibration to tritium in the aquifer from the former injection well at INTEC (DOE Idaho, 2006e).

Minimum horizontal compressive stresses strike northeast to southwest in the Eastern Snake River Plain (Smith, 2004). Anisotropy in horizontal stresses affects both the SRPA and the vadose zone by influencing hydraulic conductivity anisotropy. Optimal orientation conditions are present for dilation or shear displacement (for fractures with moderate dip in either direction) for northwest-striking vertical fractures (Moos and Barton, 1990; Jackson, et al., 1993). Other fracture orientations are under normal compression and tend to close. Southeastward flow is thus enhanced, even when the potentiometric surface encourages southwestward flow (Smith, 2004).

1.1.2.5 Existing Contamination

Knowledge regarding existing contamination at the disposal facility is integral to NRC's ability to perform its review in a risk-informed, performance-based manner. Analysis of monitoring data related to existing contamination can help reduce the uncertainty in PA model predictions and provide additional confidence that performance objectives can be met.

There are two sources of contamination in the subsurface at INTEC: (i) contamination from a former injection well that released low activity waste water directly into the aquifer south of the TFF and (ii) contamination from piping leaks. The subsurface at TFF is significantly contaminated as a result of a 1972 leak of 70,400 L [18,600 gal] of SBW which entered the vadose zone during an unsuccessful attempt at transferring the waste between tanks (see Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) waste unit CPP-31[3], Figure 6). Approximately 5.9×10^{8} MBq [1.59×10^{4}] Ci of strontium (Sr)-90 and 1.1×10^{5} MBq [3 Ci] of technetium (Tc)-99 were released during the event (DOE Idaho, 2006e). Maximum concentrations in the shallow, northern perched water underneath the TFF are currently 200,000 pCi/L for Sr-90 (order of magnitude less than imediately following the release), while concentrations in the saturated zone are currently 3,000 pCi/L for Tc-99 (DOE Idaho, 2006d; DOE Idaho, 2006e). As a basis for comparison, the maximum contaminant level (MCL) applicable to drinking water in the SRPA[4] is 8 pCi/L for Sr-90 and 900 pCi/L for Tc-99. It is important to note that the performance objectives are not based on MCLs (although under CERCLA, DOE must meet MCLs established by EPA for drinking water in the SRPA) and MCLs are only provided to give a relative indication of the contamination levels present in the groundwater at INTEC.

As stated above, Tc-99 released from TFF piping in 1972 has since migrated through the vadose zone and is currently contaminating the SRPA beneath INTEC (DOE Idaho, 2006e). Based on data analysis and modeling performed for the updated Remedial Investigation and Baseline Risk Assessment (RI/BRA) for the TFF, there appears to be a hydraulic connection between the TFF and the 114 m [380 ft] sedimentary interbed under the TFF that has allowed rapid Tc-99 transport to the aquifer near well ICP-MON-A-230 (see Figure 7) (DOE Idaho, 2006d). Another recently constructed well 450 m [1,500 ft] away from ICP-MON-A-230 shows Tc-99 plume is fairly widespread. Due to the release of radioactivity into the environment from the 1972 event and from the injection well located south of the TFF [injected low activity waste from evaporator vapors (primarily tritium (H)-3, iodine (I)-129, and other volatile or semi-volatile radionuclides) directly into the SRPA], the constituents of concern (CoCs) in the groundwater that are currently above MCLs are Tc-99, Sr-90, I-129, and nitrate. Other CoCs include H-3, neptunium (Np)-237, plutonium (Pu)-239, Pu-240, and uranium (U)-234.

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[3]All soil release sites are assigned a number to aid in tracking information for each site (e.g., CPP-31). The "CPP" prefix indicates that the site is located at INTEC.

[4]The EPA standards for drinking water are numerically equal to the Idaho groundwater quality standards.

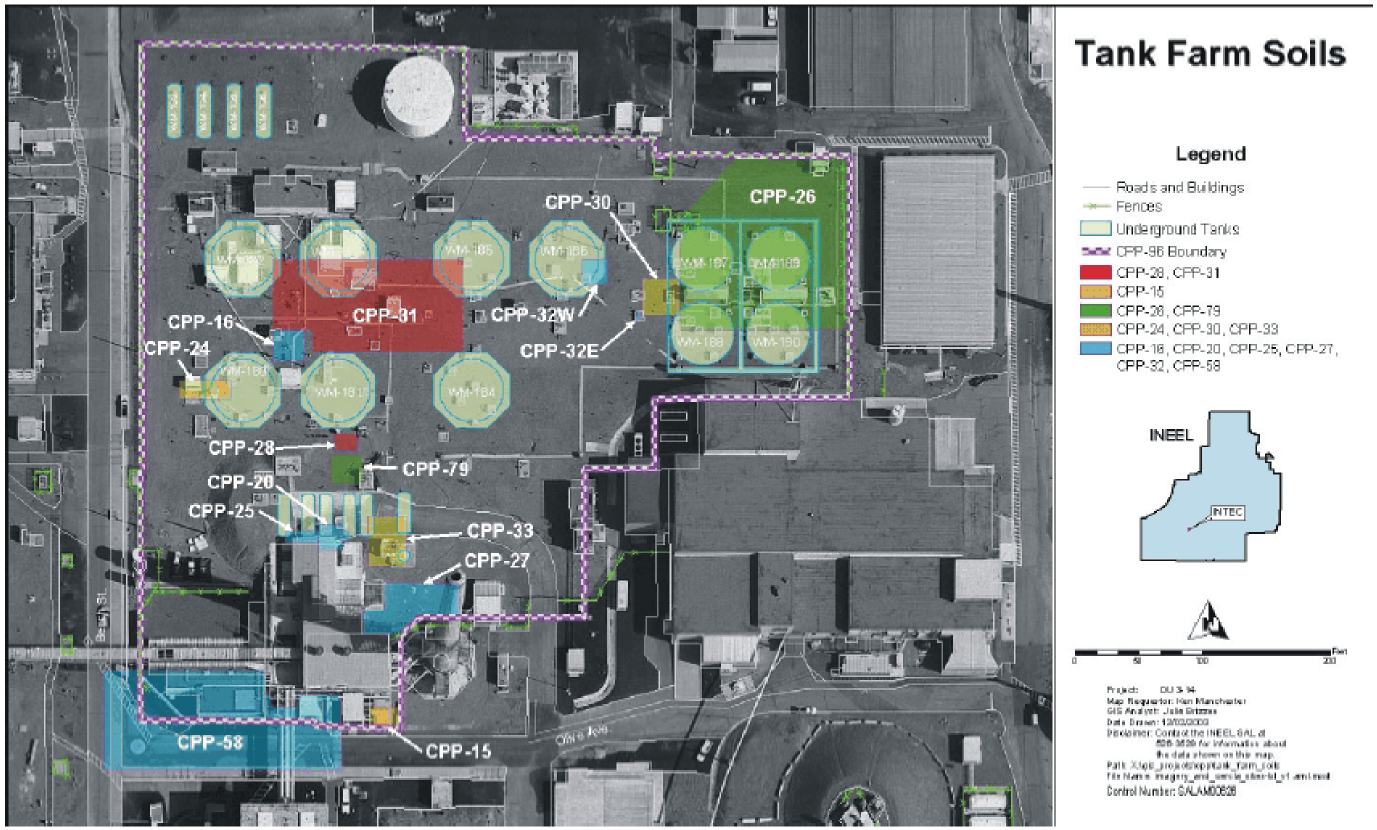

Figure 6. CPP-31 Release Site (From Figure 3-11, DOE Idaho, 2004)

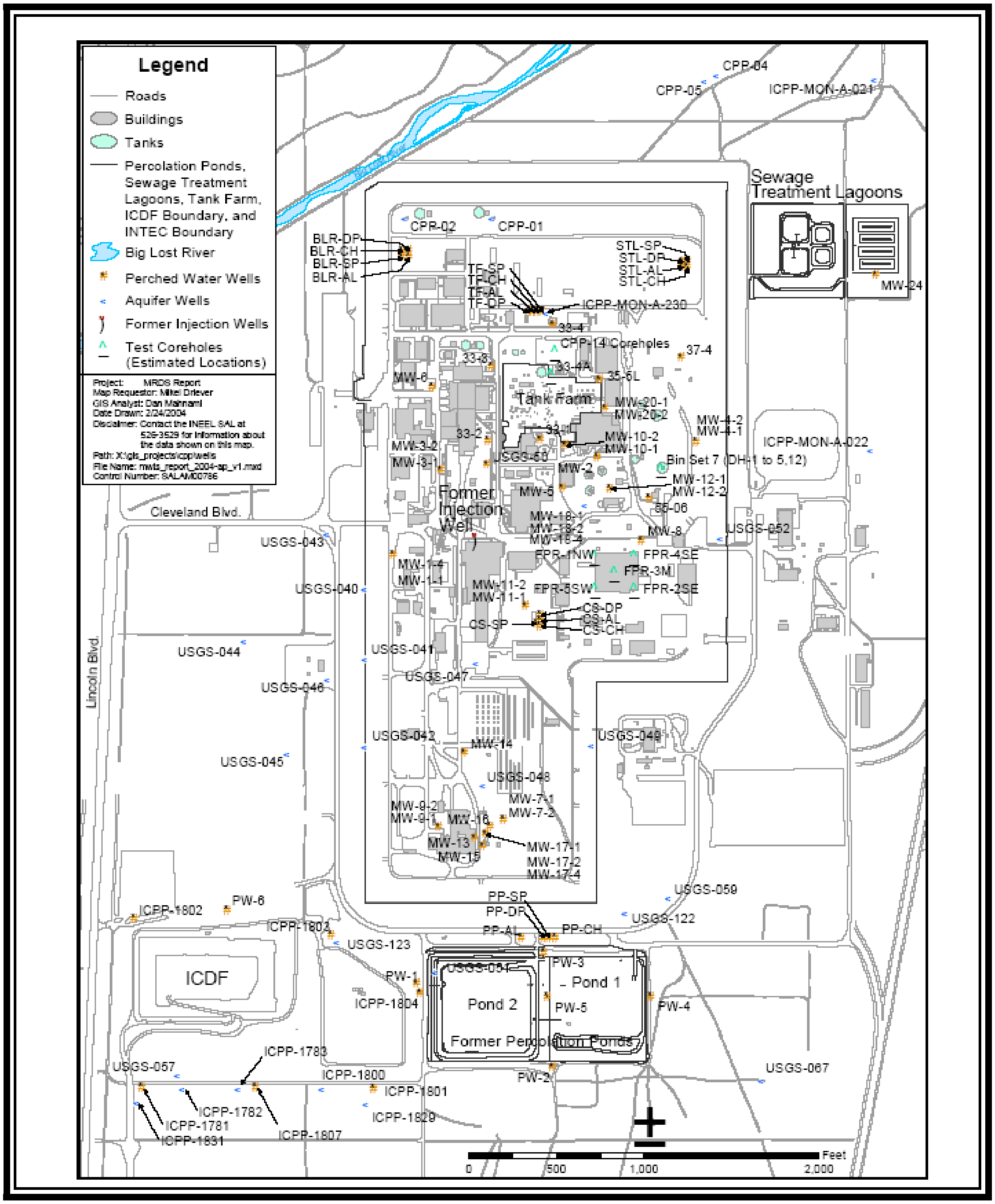

Figure 7. Monitoring Well Locations (From Figure 2-1, DOE Idaho, 2004)

A final action for contaminated perched water and an interim action for groundwater have been implemented under DOE Idaho's CERCLA program. The former INTEC percolation ponds were replaced with ponds located approximately 2 mi [3.2 km] west of INTEC. The final action for perched water also includes minimizing lawn irrigation, elimination of steam condensate discharges, closure of infiltration trenches, upgrading of surface water drainage systems, repair of leaks, potential lining of BLR, and monitoring. The interim action for the SRPA includes institutional controls to prevent current and future groundwater use until drinking water standards can be met. Action levels were established for the SRPA that could trigger treatability studies, and a "pump and treat" alternative is a contingent activity for the treatment of groundwater outside the INTEC fence. For contaminated soils, institutional controls (i.e., access controls) for workers were put in place. Infiltration controls were put in place (e.g., grading and surface sealing of soils) (DOE Idaho, 2006e). Remedial alternatives for groundwater and soils at the TFF were recently evaluated (DOE Idaho, 2006f). A proposed plan was issued in August 2006 announcing the preferred alternative (DOE Idaho, 2006g).

Using a risk-informed, performance-based approach, NRC staff will provide information in the chapters that follow that consideration of monitoring data and existing contamination provides one line of evidence that supports DOE Idaho's demonstration that NDAA criteria can be met.

1.2 DOE Idaho Tank Closure Strategy

DOE Idaho is closing the TFF tanks in response to a January 1990 Notice of Noncompliance and subsequent Consent Order (State of Idaho 1992). The Idaho Department of Health and Welfare and the EPA issued the Notice of Noncompliance to DOE Idaho because the tanks in the TFF did not meet the secondary containment requirements as set forth by Idaho Administrative Procedures Act 58.01.05.009 (40 CFR 265.193). The resulting 1992 Consent Order (and subsequent modifications) (State of Idaho 1992, 1994, 1998, 1999) required DOE Idaho to permanently cease use of the five 1,000-m^3 [300,000-gal] tanks contained in five pillar-and-panel vaults by June 30, 2003. The Consent Order also required DOE Idaho to permanently cease use of the remaining 1,000-m^3 [300,000-gal] tanks by December 31, 2012, or bring the tanks into compliance with secondary containment requirements. DOE Idaho decided to close the TFF tanks because radiation fields would make compliance with secondary containment requirements impractical, and DOE Idaho did not anticipate a need for such storage after 2012 (DOE Idaho, 2005).

The 11 1,000-m^3 [300,000-gal] underground storage tanks[5] are contained in octagonal or square concrete vaults. These tanks are stainless steel vessels with an inside diameter of 15 m [49 ft] and a wall height of 6.4 m [21 ft].[6] The tanks rest on sand pads distributed over the bottom of the concrete vaults. Eight of the 11 tanks contain stainless steel cooling coils on the floors and walls. The tops of the concrete vaults are covered with approximately 3 m [10 ft] of

[5]The four 100-m^3 [30,000-gal] underground storage tanks have been emptied and cleaned such that the residual inventory is insignificant compared to the eleven 1,000-m^3 [300,000-gal] tanks. Therefore, the 100-m^3 (30,000-gallon) tanks were assumed to be bounded by the 1,000-m^3 [300,000-gal] tanks in the PA. In addition, one of the 1,000-m^3 [300,000-gal] tanks, tank WM–190, is used as a spare and only contains a very small amount of waste {0.2 m^3 [<50 gal]} that is estimated to contain only a small amount of activity {3 terabecquerels (TBq) [80 curies (Ci)]} compared to the other 1000 m^3 (300,000-gal) tanks.

[6]Two tanks (WM–180 and WM–181) have 7.0 m [23 ft] high walls.

soil to provide radiation shielding. Figure 8 is a diagram of a belowground storage tank showing the tank, sand pad, concrete vault, and auxiliary piping.

To demonstrate that the TFF waste residuals and associated ancillary equipment at final closure will meet the NDAA Section 3116 criteria, DOE Idaho reviewed and analyzed historical waste management information, performance assessment (PA) results, and sampling and analysis results from the recent tank cleaning activities. In addition, the residual waste inventory at closure was updated by DOE Idaho to reflect the results of TFF cleaning activities (DOE Idaho, 2005).

In general, DOE Idaho's approach to close the TFF includes removing the SBW for treatment, then closing the tanks to meet NDAA Section 3116 criteria (DOE Idaho, 2005). The TFF tank system's closure process includes waste removal; cleaning of the tanks, piping, and ancillary equipment; and stabilization of the tank configuration and ancillary equipment. To complete SBW removal, DOE Idaho intends to remove as much of the remaining liquid and solid heel residue from the tanks and ancillary equipment as practical. Following waste removal from the tanks and TFF cleaning activities, DOE Idaho performs confirmatory sampling and analysis to assess the decontamination effectiveness and for waste characterization (DOE Idaho, 2005).

Some residual radioactivity that cannot be removed from the tanks by the cleaning process or other technically practical means will remain. The waste residuals are sampled and analyzed to

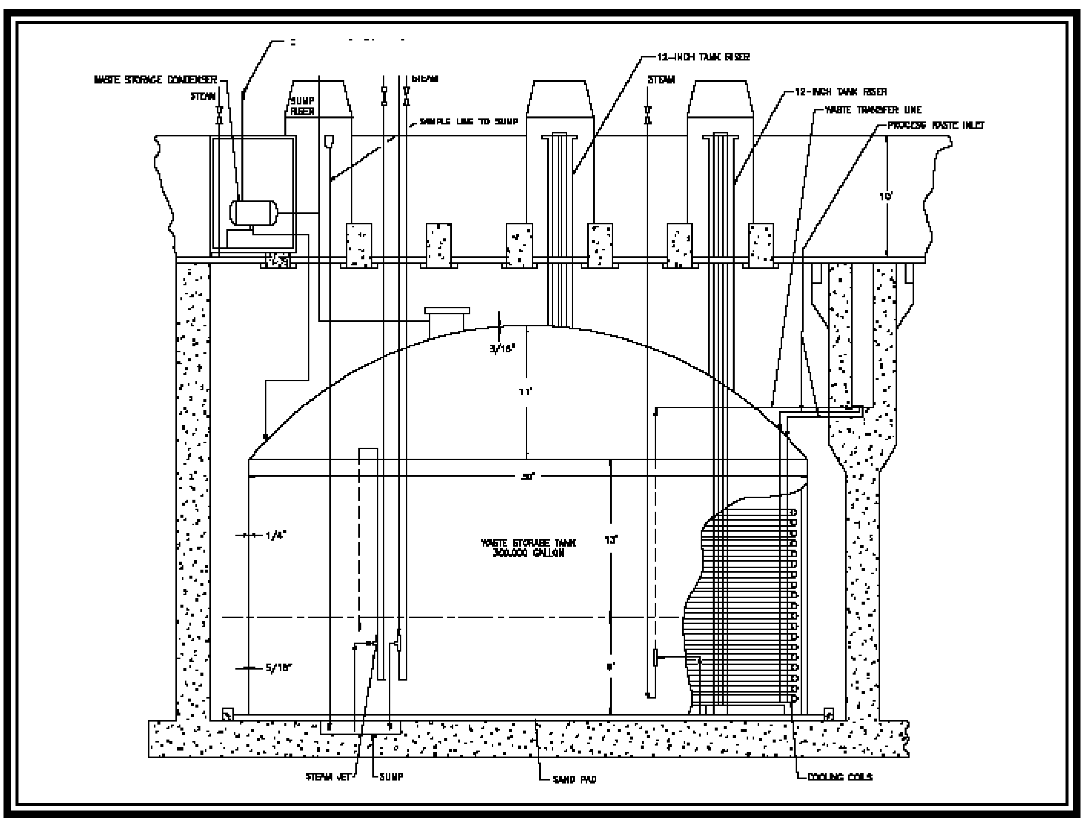

Figure 8. View of a Typical Tank/Vault System (From Figure 4, DOE Idaho, 2005)

determine the concentrations of radionuclide and hazardous constituents remaining in the tanks (DOE Idaho, 2005). For those two contaminated sand pads underlying tanks WM–185 and WM–187 where direct sampling is not practical, DOE Idaho estimated the inventory (see Section 3.1). DOE Idaho plans to stabilize the tank system by filling the system with grout. Process lines will be decontaminated and capped, and all lines (including process lines) that provide a pathway to the tanks will be grouted and capped (DOE Idaho, 2005).

A tank cleaning system (Figure 9) comprised of a washball, directional nozzle, and modified steam-jet pumping system has been developed and used thus far in the TFF tank cleaning operations (DOE Idaho, 2005). During washball and directional nozzle operations, the steam-jet ejectors remove the waste-containing slurry (solids suspended within the liquid waste) from the tank. The goal of tank cleaning is to remove as much waste as practical. During this operation, radiation levels are monitored on the steam-jet transport line to indicate cleaning effectiveness. Monitoring the radiation levels near the transport line provides the cleaning system operators and project manager information to assess when continued tank cleaning ceases to be effective. When radiation levels decrease to the lowest value [near 0 counts per minute (cpm); see Figure 10] and remain constant, cleaning is stopped and the tanks are inspected. A visual inspection via a remote-controlled camera is used to determine tank cleaning effectiveness, and then samples are collected and analyzed. During the visual inspection, residual solid heel depths are estimated by comparing the solids depths to benchmarks within the tanks, such as cooling coil support brackets and associated welds. For example, in tanks with cooling coils, the bottom weld and stainless steel bracket thickness measures 0.97 cm [0.38 in]. Knowing this thickness, the depth of waste next to these brackets can be estimated. A reflection from the stainless steel at the tank bottom is used to indicate that

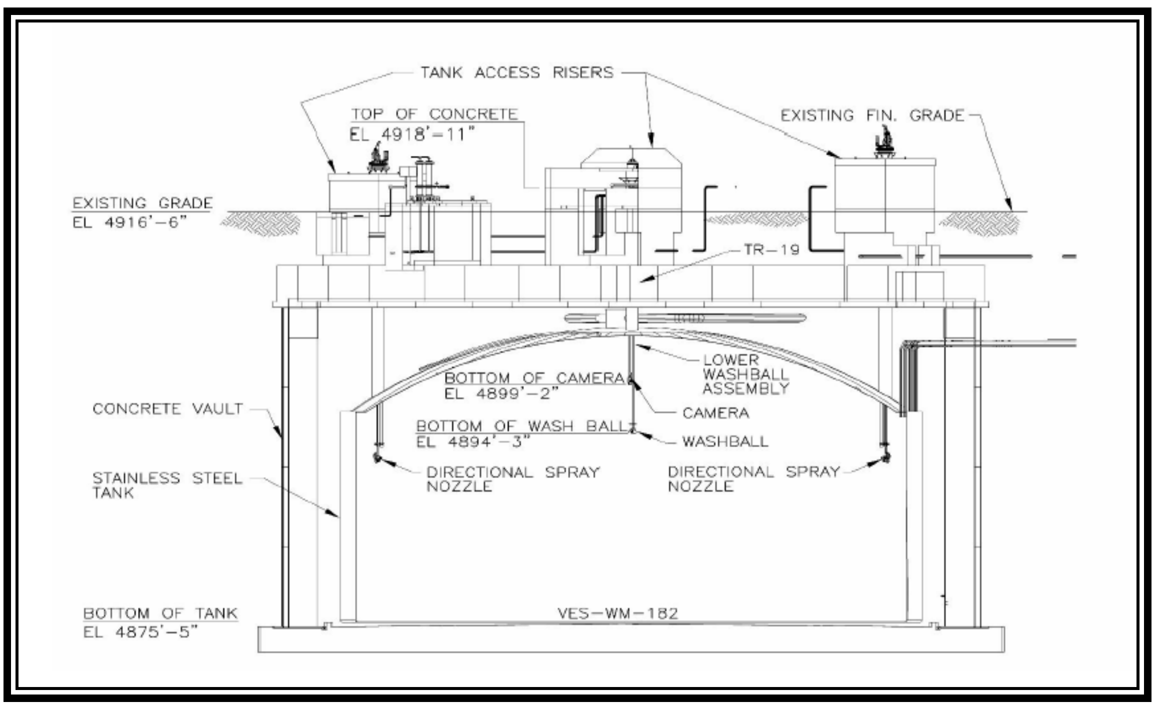

Figure 9. Typical Tank Cleaning System (From Figure 10, DOE Idaho, 2005)

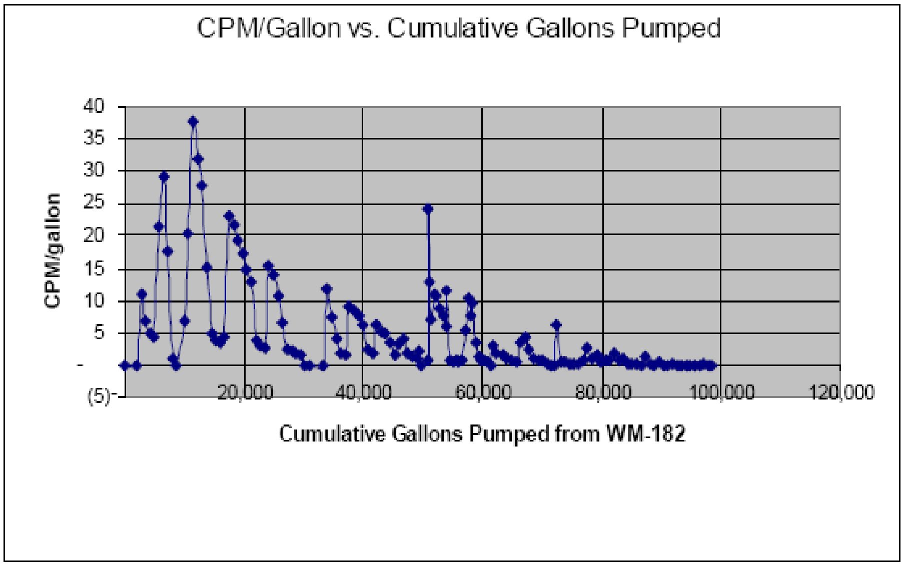

Figure 10. CPM/Gallon Versus Cumulative Gallons Pumped for Tank WM–182 (From Figure 13, DOE Idaho, 2005)

no solids are present. The radiation monitor allows tank cleaning to proceed without repeated visual inspection or sample collection and helps ensure that as much waste as practical is removed from the tanks.

Samples of the residual waste are collected with small positive-displacement pumps. Submersible pumps are lowered into the bottom of tanks or vaults through risers. The pump is activated, and liquid and solids are pumped to sample containers on the surface. The submersible pump can only reach the residual waste directly beneath the riser through which it is lowered. The residual waste is agitated before sampling.

Prior to its implementation, the TFF tank cleaning system was tested in a full-scale mockup tank using simulated waste. The washball and directional nozzle tank cleaning system and the modified steam-jet pumping system were used to slurry the solid and liquid wastes and remove them from the tanks. Steam jets were modified by cutting the steam supply and discharge lines and installing a new steam jet lower in the tank. During cleaning system development, the INL Site and DOE's Tanks Focus Area (TFA) performed a review of tank cleaning technologies which was documented in a Pacific Northwest National Laboratory (PNNL) report (2001). The TFA was formed by DOE to address all aspects of remediating radioactive wastes from underground storage tanks DOE-wide, including tank cleaning technology. This review focused on the technical feasibility and appropriateness of the approach selected by the INL Site and on technology gaps that could be addressed by using technologies or performance data available at other DOE sites and in the private sector.

The tank vaults are cleaned by iterative flushing with water. The water is removed using the existing steam jets. Process piping in the TFF is cleaned by flushing three piping system volumes through the system with a pressure equal to previous waste transfers to ensure that the pipe area contacted by waste is rinsed during the flushing operations. In all tank vaults, rainwater and snowmelt leakage through the vault roof has been pumped periodically from the vault sumps to waste tanks. Tank waste residuals remaining after cleaning and before grouting consist of a relatively small amount of solids and contaminated flush water. Prior to grouting, the small amount of liquid waste in the vault sumps will be emptied using the existing steam-jet pumps. The mockup testing shows that most of the remaining flush water and some solids will be removed during the grouting process (DOE Idaho, 2005). Grout will be used to push waste residuals toward the removal equipment (i.e., jet pumps). Any remaining residual liquid will be stabilized with a grout material (DOE Idaho, 2005). The lines connecting the vault sumps to the tanks will be grouted, followed by grouting of the vaults.

1.3 Waste Determination Criteria

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

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

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

(1) The waste does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste;

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

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

(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and

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

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

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

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

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

These are the criteria used by DOE in its draft waste determination for the closure of the INTEC TFF.

1.4 NRC Review Approach

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

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

NRC staff's review, documented in this TER, was based on DOE Idaho's "Draft Section 3116 Determination Idaho Nuclear Technology and Engineering Center Tank Farm Facility (DOE Idaho, 2005)." A publicly available version of the draft waste determination was submitted by DOE Idaho on September 7, 2005, along with approximately 140 references. The NRC staff performed a technical review of the information and sent an RAI to DOE Idaho on January 10, 2006 (NRC, 2006a). The RAI included questions about waste inventory, removal of highly radioactive radionuclides, hydrological modeling, and waste classification. In letters dated March 17 (DOE Idaho, 2006a), April 26 (DOE Idaho, 2006a), and May 31, 2006 (DOE Idaho, 2006b), DOE Idaho submitted its RAI responses and approximately 110 additional references. NRC and DOE held a public meeting on June 1, 2006, to discuss DOE's responses and had a subsequent follow on discussion during a site visit on June 20, 2006. Subsequent to these interactions, NRC staff requested that DOE Idaho provide additional information to support their waste determination and associated PA. DOE responses were submitted by four separate emails. Publicly available documents are referenced under the docket number PROJ0735 in the NRC's Agencywide Document Access and Management System (ADAMS).

NRC staff reviewed the draft waste determination and supporting documentation to assess whether it had sound technical assumptions, analysis, and conclusions with regard to meeting NDAA criteria and thus, that DOE's proposed closure of the TFF protects public health and safety and the environment. This approach is consistent with that proposed by the NRC staff in SECY-05-0073, dated April 28, 2005, and approved by the Commission in the Staff Requirements Memorandum dated June 30, 2005. This TER addresses each of the applicable criteria in the NDAA and presents the NRC staff's approach, assumptions, and conclusions, as well as identified key areas to be targeted for monitoring that are important to meeting the performance objectives in 10 CFR 61, Subpart C.

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

1.5 Previous Waste Determination Reviews for INL

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

The second WIR determination for INL concerned the same HLW tanks that are the subject of this review. The staff used the two WIR criteria provided in the NRC's Final West Valley Policy Statement (NRC, 2002a) and concluded that DOE appeared to have reasonably analyzed the relevant considerations in concluding that the residual waste in the tanks could meet the two WIR criteria (NRC, 2003b). The previous WIR review provided valuable risk insights that were used during this review, but because the NDAA criteria are slightly different and a significant amount of time has elapsed since the previous review, this TER updates the previous review to support the current INTEC TFF closure draft waste determination.

2 CRITERION ONE

The waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW (NDAA).

2.1 Waste Disposal

Criterion One allows for the consideration that waste may require disposal in a geologic repository even though the two other criteria of the NDAA may be met. Consideration could be given to those circumstances under which geologic disposal is warranted in order to protect public health and safety and the environment; for example, unique radiological characteristics of waste or non-proliferation concerns for particular types of material.

2.2 NRC Review and Conclusions

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

3 CRITERION TWO

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

The NRC staff evaluated this criterion by analyzing DOE Idaho's (i) methodology for developing radionuclide inventories for the tanks, sand pad, and auxiliary equipment; (ii) process for identifying highly radioactive radionuclides; (iii) selection of waste treatment technology; and (iv) demonstration of removal to the maximum extent practical, including analysis of the costs and benefits of additional radionuclide removal. For the purpose of reviewing DOE waste determinations, NRC staff believes that highly radioactive radionuclides (HRRs) are those radionuclides that contribute most significantly to risk to public, workers, and the environment (NRC, 2006b).

3.1 Waste Inventory and Sampling

A significant source of uncertainty in high-level waste tank closure is the concentration and volume of radionuclides in the residual materials remaining in the tanks and ancillary equipment. The inventory remaining in the tanks: (i) must be developed to demonstrate that the waste has been processed to remove HRRs to the maximum extent practical (see Section 3.7), (ii) is needed to determine if the waste is greater than Class C (see Section 4.1), and (iii) is used to develop the source term in the PA (see Sections 4.2.6 and 4.2.7). DOE Idaho used historical waste management information, coupled with post-cleaning sampling and analysis and PA results, to demonstrate that the TFF waste residuals and associated ancillary equipment will meet criterion 2 and 3 of Section 3116 at closure. DOE Idaho estimated the inventory of the large 1,000-m^3 (300,000-gal) tanks for use in its 2003 PA (DOE Idaho, 2003a) and subsequently updated its estimated inventory at closure (DOE Idaho, 2005). The following sections describe the development of the inventory and sampling of the large 1,000-m^3 [300,000-gal] and small 100-m^3 [30,000-gal] tanks, sand pads, and auxiliary equipment.

3.1.1 Large Tank Inventory

Because tank cleaning activities had not been completed when DOE Idaho initially prepared the PA to support closure activities, DOE Idaho used an estimated baseline inventory and assumed cleaning efficiency for four cases, described below, to demonstrate compliance with performance objectives (DOE Idaho, 2003a). The following assumptions were used by DOE Idaho in development of the PA inventory (DOE Idaho, 2003a):

- The highest measured radionuclide concentrations from tanks WM–182, WM–183, and WM–188, recently sampled at the time the PA for the TFF was written, were used for the worst-case inventory. Tank WM–188 had the highest radionuclide inventory out of these three tanks (see worst-case inventory description below).

- It was assumed that each tank would initially (before grouting) contain approximately 3 cm [~1 in] or 2,317 kg [5,108 lbs] of tank solids. These solids were estimated to have a bulk density of 1.4 g/cm^3 (with 25 percent solids and 75 percent free interstitial liquid by volume). In addition, each tank was assumed to contain 4,989 L [1,318 gal] of liquids, which corresponds to approximately 3.2 cm [1.3 in] of total material remaining in the bottom of the tank.

- It was assumed that radionuclide concentrations in the solid materials would be unaffected by tank cleaning and that tank cleaning would only result in limited bulk mass removal (i.e., no removal from dissolution of solids into the large quantity of "flush" water used to clean the tanks was assumed).

To account for uncertainty, the sensitivity to four different inventories was assessed in the PA calculations (worst, conservative, realistic, and best). They are generally described in DOE Idaho (2003a) as follows:

- The worst-case inventory assumed cleaning operations were ineffective.

- The conservative-case inventory assumed cleaning operations reduced solid residual mass by 10 percent and the radionuclide concentrations in the liquid phase by half.

- The realistic-case inventory assumed a 25-percent reduction in the solid residual mass and an 80-percent reduction in the radionuclide concentrations in liquid.

- The best-case inventory predicted a 50-percent reduction in solid residual mass and a 95-percent reduction in the radionuclide concentrations in liquid.

It should be noted that the reductions stated in the bullets above refer to further reductions in the tank waste inventory after the existing transfer equipment already removed as much bulk liquid and solid heel as possible prior to cleaning. In addition, the total volume of liquid waste for each of the four inventories was assumed to remain the same.

As part of the draft waste determination (DOE Idaho, 2005), the inventory of the large tanks was updated. The estimated residual waste inventory at closure was updated to reflect the results of recent TFF cleaning activities which occurred from 2002 to early 2005. Sampling and analysis plans (SAPs) were developed for liquid and solid phases for tanks WM–180, WM–181, WM–182, WM–183, WM–184, WM–185, and WM–186 (see DOE Idaho, 2005, Chapter 8 for a bulleted list of references). DOE Idaho attempted to sample all HRRs (see Section 3.3). The results of the sampling are presented in data quality assessments (see DOE Idaho, 2005, Chapter 8 for a bulleted list of references). Limited or no analytical data are available for most radionuclides; therefore, the ORIGEN2 model is used to predict inventories of these radionuclides. Input parameters to the ORIGEN2 model include (i) fuel types, based on fuel-cladding type (e.g., stainless steel, zirconium, and aluminum); (ii) cooling time; and (iii) burnup level. Model parameters are calibrated to analytical data to allow the model to estimate SBW radionuclide inventories based on a weighted average of the different fuel types processed at INTEC. Radionuclide-to-Cs-137 ratios, called the ORIGEN2 ratios, are estimated for each radionuclide to provide relative activities, which are useful for those radionuclides that are difficult to detect. The activities are normalized to Cs-137 because this radionuclide is a major constituent in INTEC waste streams and its daughter, barium (Ba)-137m, emits a high energy gamma, making it easy to detect with confidence in radioactive waste streams. Analytical ratios based on analytical data (in lieu of ORIGEN2 modeling) are also calculated. These ratios were recently updated in 2005 and presented in Appendix A of the draft waste determination (DOE Idaho, 2005) to estimate the inventories for each tank at closure using the decay-corrected ratios for the expected year of 2012 closure (Wenzel, 2005).

DOE Idaho stated it was only able to report one solid sample result due to the inability to collect enough solid material from the tanks (DOE Idaho, 2005). In its response to RAI 2, DOE Idaho stated that for solid analysis the SAPs require 15-percent solids by volume and that this requirement was not met for any of the tank sampling activities. Only the samples collected from WM–183 contained any visible solids. All of the WM–183 liquid samples were filtered to composite the solids into a single solid sample of a few grams for analysis (DOE Idaho, 2006a). The solid results for WM–183 were used to estimate the residual solid inventories for all tanks. For radionuclides that were not sampled or not detected (most of the radionuclides that are not HRRs), the remaining inventory was estimated using ORIGEN2 ratios and the mean Cs-137 concentration from sampling conducted prior to cleaning tank WM–188, because this tank had the highest pre-cleaning concentration for Cs-137. For liquid samples, the 95 percent upper confidence level of the Cs-137 liquid samples for each tank is used to estimate the concentrations of radionuclides that were not sampled in that tank. Using this approach, tank WM–182 had the highest postcleaning inventory of 2,394 Ci (DOE Idaho, 2005).

The final inventory for each tank was estimated using the concentrations obtained from sampling as discussed above. An assumed density of 1.4 g/cm^3 was used with estimated volumes of the residual solids in each tank to determine the mass of solids. With this information, the total solid heel inventory could be calculated (product of concentration and mass of solids). The tank volumes were estimated by viewing videotapes of the tanks taken before, during, and after the final cleaning and sampling events and estimating depths from reference points in the tanks (e.g., cooling coil support brackets and associated welds). Areas of the tank that showed the reflective surface of the stainless steel bottom were assumed to have a depth of contamination of 0 cm [0 in]. The thickness of the solid heel varied between 0 and 0.97 cm [0 and 0.38 in]. The residual solid heel volume was estimated using simple kriging methods (e.g., point-kriging with linear interpretation) in the computer code Surfer 8 to create the surfaces. Residual solid volume estimates were also made with Surfer 8 using a lower 0 cm [0 in] surface boundary, the kriged upper surface boundary, and the trapezoid rule to estimate the residual solid volume. Surface contour maps (see Figure 11) and estimated volumes are provided in engineering design files (EDFs) prepared for each tank (see DOE Idaho, 2005, Section 2.4.2, for a list of references). The depth of liquid above the interstitial waste was estimated to be approximately 3 cm [1 in] compared to the height of the steam-jet lines in the post-decontamination video.

DOE Idaho reported that the total postdecontamination inventory for each of the cleaned TFF tanks is significantly less than the total “conservative” post-decontamination inventory (see description above) of a single tank estimated in the PA 8.9×10^8 MBq [24,103 Ci] used for the compliance case (see Table 1). As stated above, tank WM–182 has the highest inventory with an estimated total of 8.9×10^7 MBq [2,393 Ci] in the solid and liquid residual waste combined (DOE Idaho, 2005). See Table 1 below for the final estimated inventory for a single tank (WM–182). DOE Idaho used the dose results from the PA and scaled those results to the ratio of the estimated final inventory at closure based on sampling (Table 1) to the “conservative” PA inventory (also shown in Table 1) to compare final estimates of radiological risk against performance objectives (see Section 4.2).

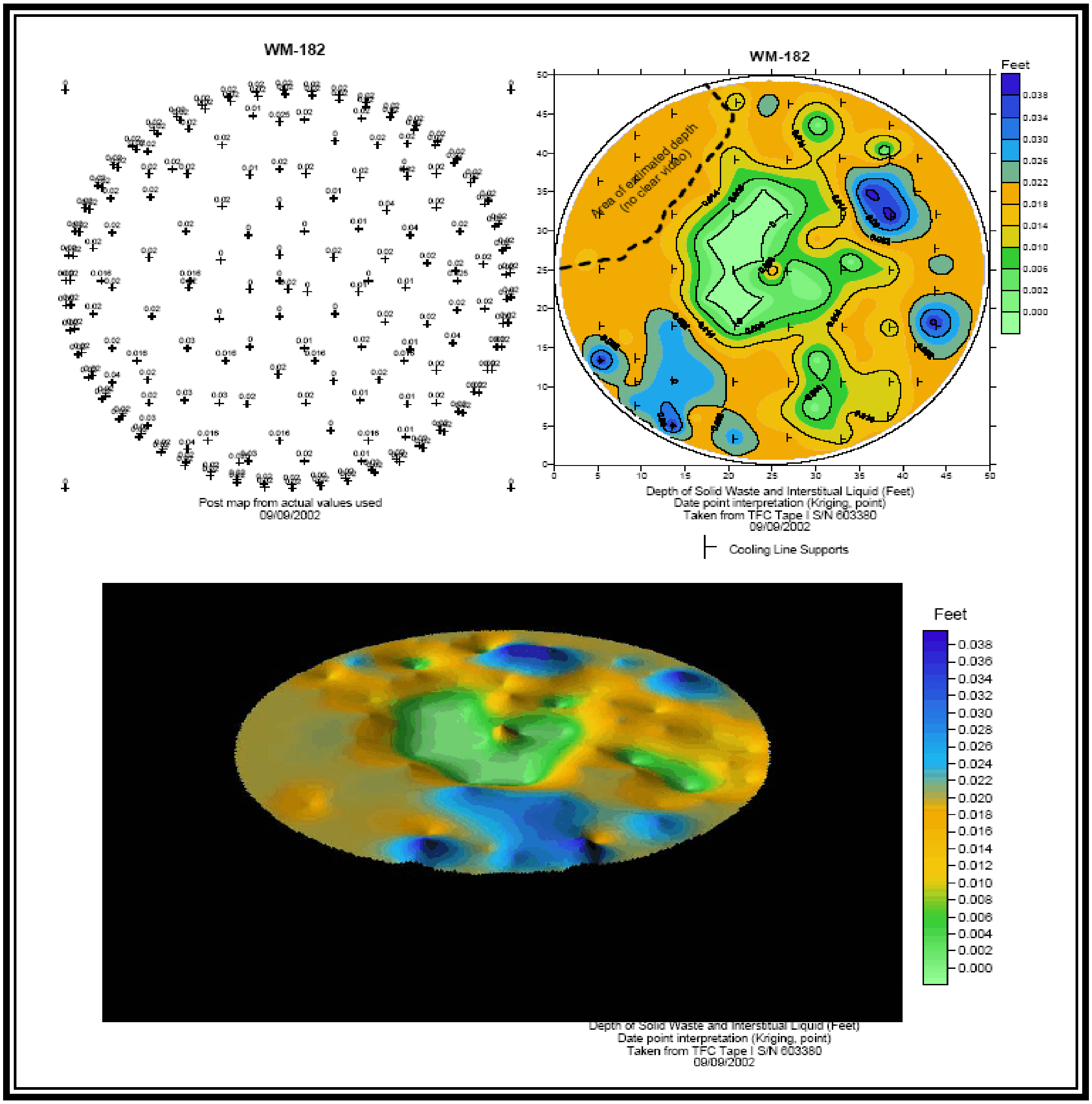

Figure 11. Kriging Analysis to Determine Residual Solid Waste Volume (From Figure 1, Portage 2005a)

Table 1. Maximum Expected Residual Waste Inventory of a Single Tank at Closure (Tank WM–182) and DOE PA (DOE Idaho, 2003a) Conservative Single Tank Inventory (modified from Table 1, DOE Idaho, 2005)

Radionuclide	**Total Residuals (Ci)***	
	DOE Waste Determination†	DOE PA
Am-241	4.2×10^{-1} §	6.2×10^{-1}
Ba-137 ‡	1.14×10^{3}	4.0×10^{3}
Cm-242	1.32×10^{-3}	1.3×10^{-3}
Cs-137	1.14×10^{3}	4.0×10^{3}
C14	4.96×10^{-6}	4.9×10^{-1}
Co-60 ‖	6.3×10^{-2}	2.1×10^{-1}
I-129	7.74×10^{-4}	2.6×10^{-3}
H-3	7.17×10^{-1}	5.6×10^{-1}
Nb-94	2.06×10^{-1}	7.7
Ni-59	2.51×10^{-2}	1.7×10^{-1} ^
Ni-63	2.86	3.0
Np-237	4.70×10^{-2}	7.6×10^{-3}
Pu-238	1.14×10^{1}	1.7×10^{1}
Pu-239	3.4	1.2
Pu-240	1.35	1.1
Pu-241	1.95×10^{1}	1.5×10^{1}
Pu-242	9.88×10^{-4}	8.4×10^{-4}
Sr-90	2.34×10^{1}	8.0×10^{3}
Tc-99	7.64×10^{-1}	1.0
Y-90	2.34×10^{1}	8.0×10^{3}
Total #,** Ci all radionuclides	2.4×10^{3}	2.41×10^{4}

*Ci = 3.7×10^{4} MBq.

†Draft waste determination radionuclide inventories are based on (i) waste residuals estimated using remote video inspection of cleaned tank internals to map out estimates of depth of remaining residual solids and liquids across tank bottoms using tank internal reference points of known height, (ii) best estimated radionuclide concentrations from past and recent samples, and (iii) radioactive decay to 2012. Analytical results for tank WM–182 were used to calculate the liquids inventory at closure for Am-241, C-14, Cs-137, Eu-154, H-3, I-129, Np-237, Pu-238, Pu-239, Pu-241, Sb-125, Sr-90, Tc-99, U-234, and Y-90. Analytical results for Tank WM-183 were used to calculate the solids inventory at closure for Am-241, Ba-137m, Co-60, Cs-137, I-129, Nb-94, Pu-238, Pu-239, Sb-125, Sr-90, Tc-99, U-234, and Y-90.

‡A 1:1 ratio is assumed for Cs-137 to Ba-137m as a conservative estimate of radionuclide inventory; based on decay probability, Ba-137m is approximately 95 percent of the Cs-137 inventory.

§The value of 0.42 Ci was taken from Table A–7 and A–12 of DOE Idaho (2005). The value in Table 1 of DOE Idaho (2005) appears to be a typographical error.

‖Co-60 was not listed in Table 1 of the draft waste determination (DOE Idaho, 2005); however, it is listed in Table 2 of 10 CFR 61.55 and is therefore, an HRR by default (using DOE Idaho's methodology). It is also included on DOE Idaho's Table 5 of HRRs (2005).

^The DOE PA inventory value for Ni-59 reported in Table A–12 of the draft waste determination (DOE Idaho, 2005) of 0.04 is different than the value reported in Table 2-16 of the DOE PA (2003a).

#Radionuclides shown are contributors in PA dose calculations or regulated by concentration limits in 10 CFR 61.55. The waste determination total is based on the entire inventory of radionuclides.

**The DOE PA inventory total is based on the entire inventory of radionuclides decayed to the year 2016 [the expected date of closure when the DOE Idaho PA (2003a) was written].

3.1.2 Small Tank Inventory

An inventory was calculated by DOE Idaho for the 100-m^3 [30,000-gal] tanks and presented in the PA. A comparison of the activity calculated for the 100-m^3 [30,000-gal] tanks with the activity in the 1,000-m^3 [300,000-gal] tanks indicates that the contamination levels in the 1,000-m^3 [30,000-gal] tanks are insignificant. DOE Idaho reasoned that the inventory for the 1,000-m^3 [300,000-gal] tanks would bound any releases from the 100-m^3 [30,000-gal] tanks (DOE Idaho, 2003a).

The 100-m^3 [30,000-gal] tank inventory was updated in the Draft Section 3116 Determination document (DOE Idaho, 2005). The inventory was based on analytical data from postcleaning sampling. Solid residual samples were not collected because an adequate volume of material was not present in the tanks. A film layer was observed on the lower half of all four tanks that appeared to be algae or another form of biological growth and was not likely to contain any significant radioactivity. However, to establish a conservative estimate, DOE Idaho assigned the film layer a thickness of 5 mils [0.005 in] and used solid sample concentrations from tank WM–183 to estimate solid inventories. The inventories for each 100-m^3 [30,000-gal] tank vary from 36.2 to 36.7 Ci. Tank WM–106 has the highest remaining Ci content of the 100-m^3 [30,000-gal] tanks. Because the inventory of the 100-m^3 [30,000-gal] tanks is significantly lower (two orders of magnitude) than the 100-m^3 [300,000-gal] tanks, the focus of this TER is on the 100-m^3 [300,000-gal] tanks.

3.1.3 Sand Pads

Tanks WM–182 through WM–190 rest on a 15 cm [6 in] layer of commercial grade sand overlying a concrete slab approximately 0.76 m [2.5 ft] thick. A 15 by 15 cm [6 by 6 in] concrete curb (or dike) encloses the sand pad area (see figure 12a and 12b) (DOE Idaho, 2005). The sand pads underlying two of the tanks (WM–185 and WM–187) were contaminated with first-cycle extraction wastes in 1962 as a result of back-siphoning events. The waste entered the tank vault sumps and was pumped back into the tanks approximately 24 hours later (DOE Idaho, 2003a). Because the first-cycle extraction waste in the vault sump overtopped the vault sump and curb (or dike) which holds the sand in the sand pad, radionuclides were expected to be transported radially into the sand pads underneath the tanks (see Figure 12a and 12b). Before and after these releases, water from precipitation, spring runoff, and irrigation infiltrated the tank vaults to the sumps and sand pads and was pumped out at least semi-annually, removing contaminants from the sand pad in liquids that drained to the vault sump through drain tubes in the curb (or dike). The residual inventory in the sand pads was developed by DOE based on an analytical, one-dimensional diffusion model with thirty-eight pumping events (pumping of contaminated water that drains from the sand pad into the vault sump). The actual number of water transfers from the sand pad likely exceeds 130 for each vault to date (DOE Idaho, 2003a). The sand pad was modeled to fill with water; radionuclides partition from the contaminated solid sand particles to the aqueous phase and are subsequently removed when the sump pump is operated. No direct sampling of radionuclide concentrations in the sand pad has been performed to date. Indirect sampling of vault sumps associated with cleaned tanks was performed in accordance with SAPs (see DOE Idaho, 2005, Chapter 8, for a list of references). This data was analyzed (DOE Idaho, 2006a) in response to RAI 1 (NRC, 2006a). In the absence of direct sampling, sand pad inventories have been difficult to evaluate with confidence. DOE Idaho calculations rely on a number of assumptions regarding the initial inventory after the 1962 event and incremental removal over a 38-year period to the year 2000,

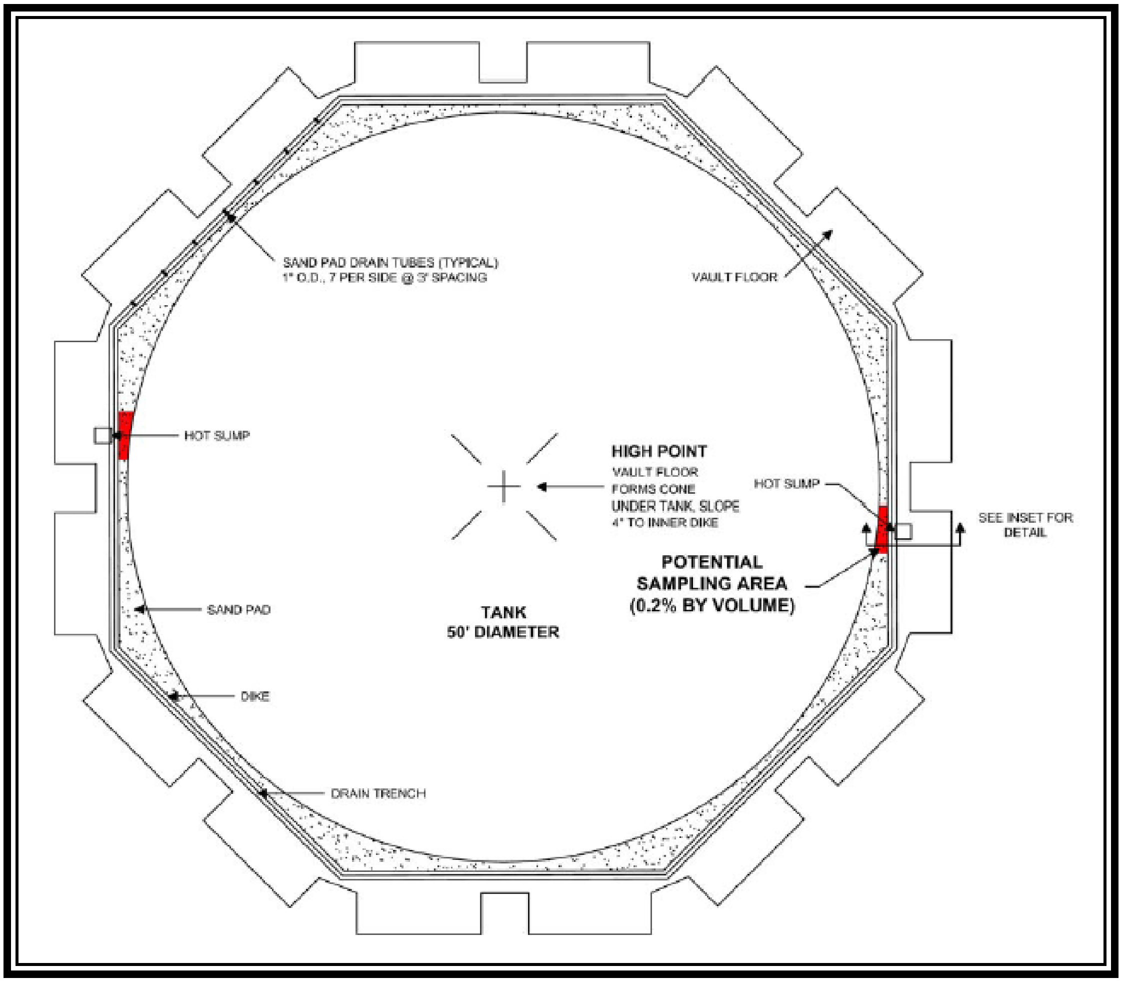

Figure 12a. Plan View of Tank, Vault, and Sand Pad (From DOE Idaho, 2006a)

with 12 additional years of decay from 2000 to 2012 (the expected year of closure). Additional sensitivity analysis was performed by DOE Idaho (2006a) to address the uncertainties with the sand pad inventory (addressed in RAIs 3 and 4 of NRC, 2006a), which could dominate the predicted dose from short-lived radionuclides assumed by DOE Idaho to be released from the vaults after 100 years, prior to substantial decay. The corrected sand pad inventory is presented in Table 2 (the sand pad inventory in Table 3 of DOE Idaho, 2005, has incorrect values, which were corrected in response to clarifying RAI 19, DOE Idaho, 2006a).

3.1.4 Auxiliary Equipment

In addition to residual waste remaining in the tanks and sand pads, process piping contains residual waste. The inventory calculated for the contaminated piping in the PA (DOE Idaho, 2003a) was updated for the Draft Section 3116 Determination based on post-cleaning sampling data.

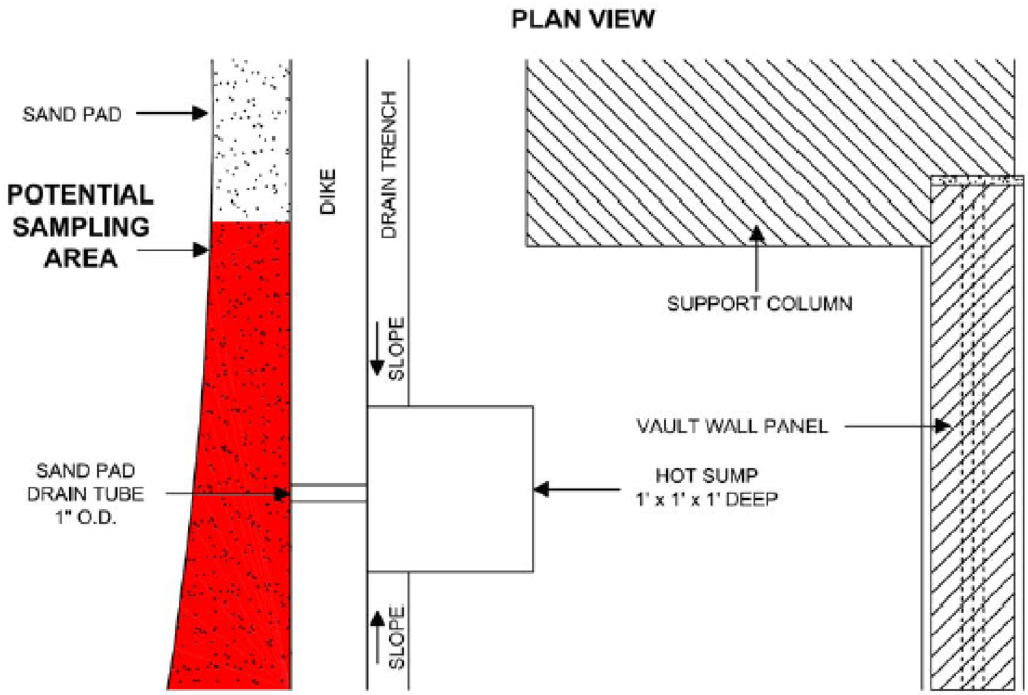

Figure RAI-3-A-2. Detailed plain view of sandpad and vault.

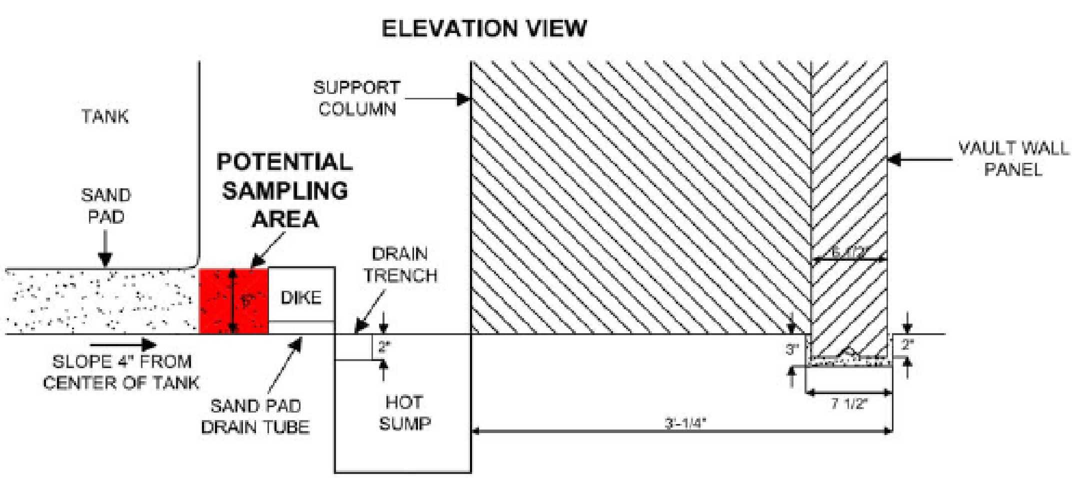

Figure 12b. Detailed View and Cross Section of Sand Pad and Vault (From DOE Idaho, 2006a)

Table 2. Sand Pad Residual Waste Inventory at Closure (Ci Per Sand Pad) (Modified From Table CR–19–1, DOE Idaho, 2006a)	
Radionuclides	**(Ci)*†**
Am-241‡	1.89
Ba-137m	1.65×10^{3}
Cm-242	1.38×10^{-5}
Co-60§	2.7×10^{-3}
Cs-137‡	1.65×10^{3}
C-14‡	3.90×10^{-7}
I-129‡	1.08×10^{-6}
H-3	3.10×10^{-22}
Nb-94	2.29×10^{-2}
Ni-63	1.69×10^{-10}
Np-237	3.72×10^{-4}
Pu238‡	2.06
Pu-239‡	1.57
Pu-240‡	3.54×10^{-1}
Pu-241	2.28
Pu-242	5.69×10^{-5}
Sr-90‡	2.49×10^{2}
Tc-99‡	2.02×10^{-12}
Y-90	2.49×10^{2}
Total Ci‖	3.85×10^{3}

*1Ci = 3.7×10^{4} MBq.
†Radioactive decay to 2012.
‡Radionuclides based on results from the PA (DOE Idaho, 2003a).
§Co-60 was not listed in Table 3 of the draft waste determination (DOE Idaho, 2005); however, it is listed in Table 2 of 10 CFR 61.55 and is therefore, an HRR by default (using DOE Idaho's methodology). It is also included on DOE Idaho's Table 5 of HRRs (2005). The value of 1.6×10^{-3} Ci for 2016 (DOE Idaho 2003a) was adjusted to 2012.
‖Radionuclides shown are contributors in DOE PA dose calculations or regulated by concentration limits in10 CFR 61.55. The total is based on the entire inventory of radionuclides, not just the key highly radioactive radionuclides presented in this table.

Estimates of the waste remaining in the process piping were developed from characterization of process waste piping sections associated with tank WM–182 that were decontaminated and removed from the system. To account for the uncertainty in how the data were collected and the limited amount of piping sampled, DOE Idaho stated that a safety factor of 500 was applied to the piping inventory (DOE Idaho, 2005). As discussed in response to NRC's clarifying RAI 12, the inventory remaining in the process piping is very small compared to the residual waste remaining in the tanks and sand pad and is insignificant with respect to the sum of the fractions calculations for waste classification (DOE Idaho, 2006a). DOE Idaho explained that the safety factor of 500 was used to provide a bounding estimate for transfer piping, valve boxes, and piping encasements in the absence of data and acceptable best practices to evaluate the auxiliary equipment. The safety factor was expected to address the uncertainty in the (i) sampling method which used metals as a surrogate for radionuclides; (ii) concentrations which did not consider areas where radionuclides could accumulate or be present in higher concentrations (e.g., bends in piping and areas near valves); and (iii) lack of consideration of fixed contamination in the inventory calculations (DOE Idaho, 2006).

The total inventory in the piping was estimated to be approximately 1.11×10^6 MBq [30 Ci] (DOE Idaho, 2005), significantly less than the piping inventory established in the PA (DOE Idaho, 2003a). However, the concentration of plutonium (Pu) isotopes important to waste classification actually increased. The piping inventory is important for certain intruder scenarios (see Sections 4.2.14 and 4.2.15); however, the predicted doses from the piping are expected to be insignificant with respect to the groundwater pathways (see Sections 4.2.12 and 4.2.13) and to the overall activity present at the TFF for demonstrating HRR removal to the maximum extent practical (see Sections 3.7 and 3.8).

3.2 NRC Review and Conclusions—Waste Inventory and Sampling

The NRC staff had many concerns associated with tank and sand pad inventory development for the TFF. One of the most significant concerns was with the lack of solid sample results for the cleaned tanks. Most of the radionuclide inventory is associated with the solid residual waste in the cleaned tanks. As stated in the Draft Section 3116 Determination (DOE Idaho, 2005), a SAP was developed for each cleaned tank that used the data quality objective (DQO) process to determine the sampling strategy, number of samples, and analytical methods to be used during sampling. Issues such as the representativeness of the samples obtained and the homogeneity of the population sampled were addressed in these SAPs. The DQO process is used to design sampling plans to support decisionmaking. The DQO process helps ensure that the type, quantity, and quality of data used in decision-making are appropriate for the intended application of the data. The analytical results for each cleaned tank have been reported in a series of data quality assessment (DQA) reports (see DOE Idaho, 2005, Chapter 8 for a list of references). A DQA is a scientific and statistical evaluation of the quality of the data to determine whether the data meet the DQOs established for sampling (DOE Idaho, 2005). Because almost the entire inventory for the TFF is based on one solid sample (e.g., WM–182 estimated solid inventory is 8.8×10^7 MBq [2,391 Ci] versus 1.1×10^5 MBq [3 Ci] in the liquid phase), NRC staff developed RAI 2 regarding the adequacy of this one solid sample (NRC, 2006a). The impact of sampling on evaluating compliance with Section 3116 criteria is discussed in detail in sections that follow.

3.2.1 Evaluation of Use of ORIGEN2 Ratios

DOE Idaho addressed NRC staff’s concern regarding the use of ORIGEN2 ratios to estimate the concentrations of radionuclides not sampled or not detected in the post-cleaning sampling (DOE Idaho, 2006a). Because approximately 98 percent of the estimated activity is based on analytical results, the impact of using ORIGEN2 ratios is minimized. The apparent overestimation of removal efficiency in the PA (DOE Idaho, 2003a) for certain HRRs in many cases was actually a result of updated ORIGEN2 ratios or sampling in 2005 which increased the relative concentrations of these radionuclides. Using ORIGEN2-based models designed for liquid and calcined waste to estimate the activity of the TFF solids results in comparatively large uncertainties, because the solid residuals are not derived directly from the TFF liquid SBW, nor do they have the same relative amounts of various constituents. There is a high degree of uncertainty in the inventory estimation when the relative activity of radionuclides in the undissolved solids is assumed to be the same as in the liquid waste (Millet, et al., 2005). Thus, use of ORIGEN2 ratios for estimating the solid residual inventory of HRRs is not recommended in the future. For the uncleaned tanks, DOE Idaho should continue to sample HRRs to ensure adequate inventory estimates for the purposes of demonstrating compliance with the NDAA criteria. For the cleaned tanks, there is no apparent negative impact of using ORIGEN2 ratios for those radionuclides not sampled for the purpose of evaluation of NDAA Criterion 2; waste classification; and demonstration of compliance with performance objectives because analytical data were collected for most of the HRRs in developing the final inventory. The methodology used to calculate the postcleaning inventory is, therefore, reasonable.

3.2.2 Evaluation of Interpolation Methods to Estimate Solid Volumes

In its response to NRC RAI 5, DOE Idaho provided additional information regarding the number and distribution of control points used in the kriging analysis for tanks with cooling coils and those without cooling coils (DOE Idaho, 2006a). For tanks where cooling coils could not be used for estimating residual waste thickness, DOE Idaho states that 1-in-diameter piping sections were cut into 1-in lengths, placed in the tanks, and used with the bottom tank welds to estimate solid residual depths. The approach used to map out reflective areas of the tank where no residual solids appear to exist is reasonable and prevents gross under or overestimation of residual solid volumes (see Figure 13). The apparent tendency to overstate the depth of contamination where reference points do exist in the tank appears reasonable to compensate for any potential underestimation associated with an inadequate number or distribution of interpolation points between control points in the kriging analysis. Because the density of the residual solids is fairly uncertain [i.e., dry solid samples have a density around 2 g/cm^3 (DOE Idaho, 2006a)], the use of a density of 1.4 g/cm^3 is considered realistically conservative, because DOE Idaho did not attempt to take credit for a reduction in the inventory due to the presence of interstitial liquids in the solid residual heels.

3.2.3 Evaluation of Estimated Tank Inventory for Demonstrating Removal of HRRs to the Maximum Extent Practical

In response to NRC's RAI 2, DOE Idaho explained that no meaningful statistical evaluation could be performed to prepare a DQA for the solid sampling (DOE Idaho, 2006a). The Draft Section 3116 Determination did not discuss the very significant differences between expected versus actual inventory for specific radionuclides presented in Appendix A of the document (DOE Idaho, 2005). However, DOE Idaho presented a number of additional solid sample results from various pre- and post-cleaning sampling events in tabular format for easy comparison (DOE Idaho, 2006a). Based on a review of this limited information, it appears that there is significant variability (an order of magnitude or higher) in the solid concentrations from tank to tank (e.g., Sr-90 and Cs-137). There is also significant variability in concentration from pre- and post-cleaning sampling for certain radionuclides (e.g., Sr-90 concentrations vary over an order of magnitude). The variability in concentrations of HRRs between tanks is important when estimating the expected removal efficiency of HRRs for tanks that have not been cleaned, because uncleaned tank WM–188 appears to have a significantly higher concentration of the relatively insoluble constituent Cs-137 that is present in cleaned tanks.

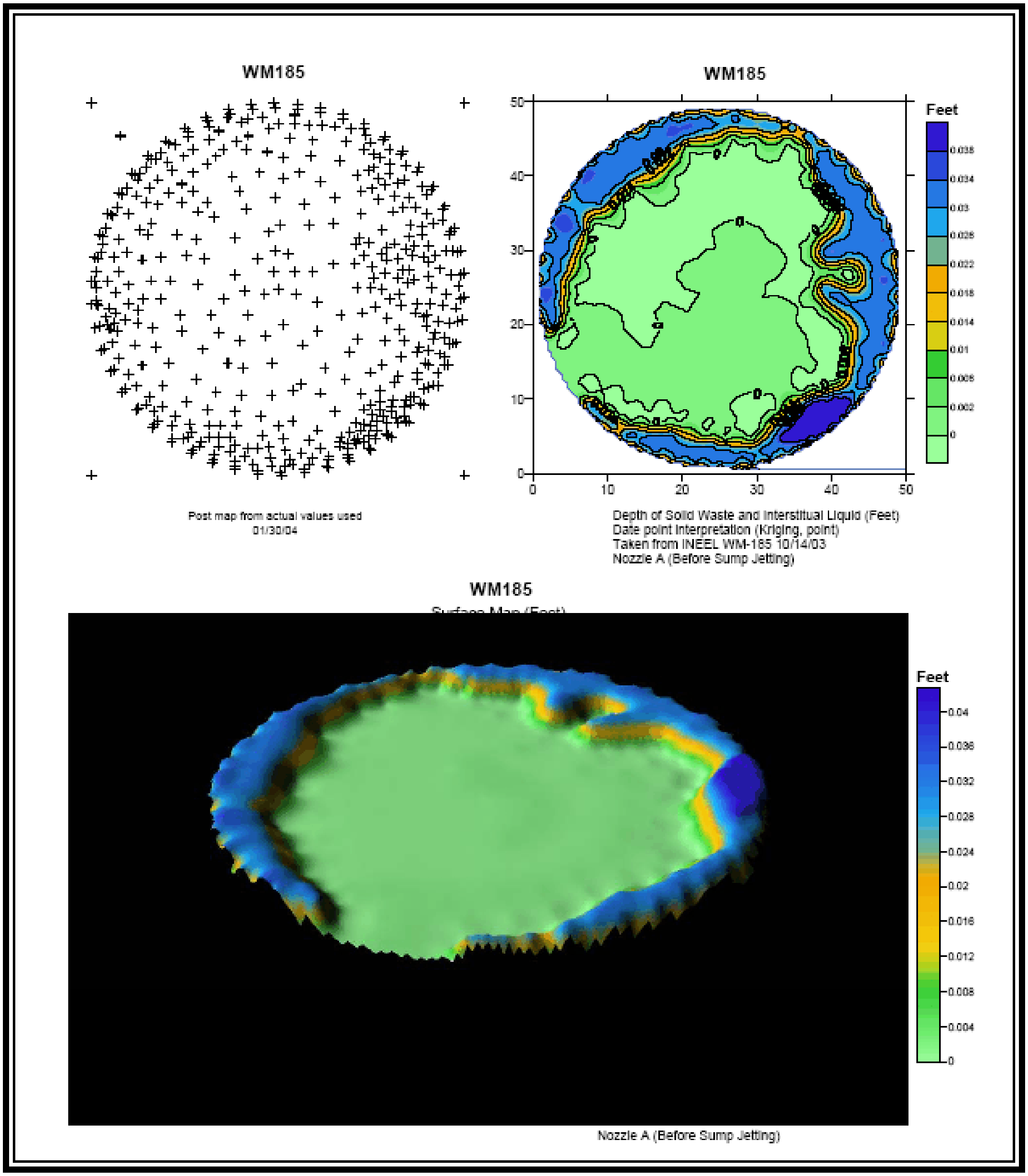

Figure 13. Surface Contour Map for WM–185 Showing Residual Contamination on Edge of Tank (From Figure 1, Portage, 2005b)

Additionally, tank WM–187 is the holding tank for waste removed from tanks WM–180 through WM–186, so there may be additional challenges with cleaning tank WM–187 due to the accumulation of solids from multiple tanks. DOE Idaho should attempt to sample tanks WM–187 through WM–190 (particularly tanks WM–187 and WM–188) following cleaning operations to ensure that the inventory for these tanks is not significantly underestimated.

3.2.4 Evaluation of Estimated Tank Inventory for Waste Classification Calculations

As stated in response to NRC staff's RAI 2, the concentrations for insoluble radionuclides (e.g., Pu-238, Pu-239, Co-60, and Am-241), appear to be fairly consistent between precleaning and postcleaning sampling events (DOE Idaho, 2006a). Furthermore, additional data for tank WM–183 collected after DOE Idaho submitted the Draft Section 3116 Determination shows little variability between the two postcleaning samples with the exception of Tc-99. These data provide additional confidence in the solid sample used to estimate the final inventory for the cleaned tanks. In response to NRC staff's RAI 17, DOE Idaho demonstrated that for highly radioactive radionuclides, the solid samples are quite comparable (DOE Idaho, 2006a) with respect to the sum of fractions calculations for waste classification (e.g., Pu-238 and 239). DOE Idaho identified (with the exception of Cm-244) and provided the inventory for each radionuclide identified in 10 CFR 61.55 that is necessary to review DOE Idaho's waste classification (see Section 4.1). Thus, the bounding inventory developed for tank WM–182 used for the purposes of waste classification is also supported.

3.2.5 Evaluation of Estimated Tank Inventory for Establishing Compliance with Performance Objectives

The NRC staff also concludes that the methodology used to estimate the tank inventory in the PA (DOE Idaho, 2003a) appears to be adequate for the purpose of demonstrating compliance with the performance objectives. However, the use of the tank WM–182 postcleaning inventory scaled to the PA (DOE Idaho, 2003a) bounding tank inventory to predict doses for the intruder scenario may underestimate the predicted dose. For example, Cs-137 is a major dose contributor for intruder scenarios, and concentrations for relatively insoluble Cs-137 are much higher in uncleaned tank WM–188 than in cleaned tank WM–182. An inventory calculated with the tank WM–188 precleaning concentrations, scaled to an expected residual solid volume remaining in tank WM–188 after cleaning, would be a more appropriate inventory to estimate the bounding intruder doses rather than adjusting the estimated dose to predict an updated intruder dose with the expectation that WM–188 will be cleaned as effectively as WM–182 (DOE Idaho, 2005). Once tanks WM–187 through WM–190 are cleaned and sampled, DOE Idaho should evaluate whether intruder dose predictions reported in its waste determination (DOE Idaho, 2005) are bounding.

3.2.6 Evaluation of Estimated Sand Pad Inventory

The NRC staff was also concerned about the lack of direct sampling of the sand pad and the fairly uncertain modeling approach and parameters used to estimate the sand pad inventory (RAIs 1, 3, and 4; NRC, 2006a). Based on information provided in the contamination event report (Latchum, et al., 1962), NRC staff questioned (NRC, 2006a) the conceptual modeling approach (e.g. one-dimensional, diffusion model) used in the PA (DOE Idaho, 2003a). In response, DOE Idaho performed additional modeling to quantify the uncertainty in the inventory for the sand pad.

In response to RAI 1, DOE (DOE Idaho, 2006a) addressed the issue regarding the lack of direct sampling of the sand pad by analyzing radionuclide concentrations of liquids sampled after vault cleaning. Liquids in the sand pad drain through drain tubes to the vault sumps. Therefore, liquid vault sump concentrations can provide an indirect method of "sampling" the liquid concentrations in the sand pad. Vaults with contaminated sand pads did not show elevated concentrations of radionuclides compared to vaults without sand pad contamination. DOE Idaho concluded that the detection of lower concentrations of most radionuclides indicates contamination in the sand pads is considerably lower than the estimated 3,850 Ci (DOE Idaho, 2006a).

Waste leakage from valve boxes and piping encasements drains to the vault sump. Therefore, DOE Idaho's conclusion is only supported if DOE Idaho calculated (from modeling) expected aqueous-phase concentrations of HRRs in the sand pad and found these concentrations to be higher than the liquid vault concentrations determined from sampling. If, on the other hand, the vault contamination is significantly higher, then any contribution from the sand pad contamination would be masked by the vault contamination from other sources. In fact, for most HRRs the vault liquid concentrations from sampling of the vault sump are significantly higher than the expected equilibrated sand pad liquid concentrations. Therefore, DOE Idaho's analysis of vault samples to support the conservatism of the sand pad inventory is inconclusive. DOE Idaho also made comparisons between Sr-90 and Cs-137 concentrations in a vault sump with a contaminated sand pad. DOE Idaho stated that higher concentrations of Sr-90 compared to Cs-137 are expected in a vault sump with a contaminated sand pad because Sr-90 is much more mobile. However, by calculating the expected Cs-137 and Sr-90 liquid concentrations using the sand pad inventory estimates DOE Idaho provided in response to NRC staff's RAI 4 (DOE Idaho, 2006a) and a simple solid-to-liquid K_d model, Cs-137 liquid concentrations would be equal to or greater than Sr-90 concentrations measured in samples collected from the sump consistent with the data presented in Figure RAI 1–A–1 (DOE Idaho, 2006a). Because the Cs-137 inventory was initially higher and much less Cs-137 is removed following each jet pumping campaign (about $1x10^{-3}$ versus $1x10^{-2}$), the relative concentration of Cs-137 to Sr-90 increases in the pore water over time and the actual liquid concentrations of Cs-137 should currently be higher than Sr-90.

Nevertheless, specifically for Sr-90, the most important potential risk driver for the sand pad considering both the uncertainty and potential magnitude of this radionuclide contribution to the peak dose (see additional discussion in Chapter 4), the observed concentrations in the vault are significantly lower than would be expected if the inventory in the sand pad were converted to an equilibrated aqueous phase concentration (assuming the sorption coefficients selected are reasonable). Therefore, there is some confidence for at least one important HRR that the inventory in the sand pad is bounding. The other important HRRs for the groundwater pathway, Tc-99 and I-129, are present in insignificant concentrations in the sand pad; therefore, the uncertainty in the inventory and concentrations for these radionuclides does not drive the risk for this source.

It is significant to note, however, that DOE Idaho's indirect method of evaluating the sand pad inventory is fairly uncertain. Uncontaminated cleaning water was used to clean the vaults prior to sampling, and the vault liquid concentrations may not be representative of the liquid concentrations from the sand pad. Therefore, the vault sampling data should only be used as an indicator regarding the conservatism of the estimated Sr-90 concentration in the sand pad. Furthermore, the additional sand pad modeling shows that the Sr-90 inventory can be eight times higher than the conservative or compliance case inventory assumed in the DOE PA (DOE Idaho, 2003a).

With respect to the sand pad inventory modeling, DOE Idaho stated in response to NRC staff's clarifying RAI 20 that over 100 "flushing events" have occurred, but only 38 events removing contamination from the sand pad were assumed. Thus, with respect to the sand pad modeling, a factor of two lower concentrations of Sr-90 would have been estimated had DOE Idaho taken credit for 100 pumping events. On balance, DOE Idaho's modeling approach attempts to bound the inventory of HRRs in the sand pad.

DOE Idaho also discussed (DOE Idaho, 2006a) the difficulty in direct sampling of the sand pads in its response to NRC staff's RAI 1. Access to the sand pads is through two 12-inch risers, which presently contain other equipment that almost completely occupies the available space. The access risers might allow collection of sand from the edge of the sand pads near the sump. Figure RAI 1–B–1 (DOE Idaho, 2006a) shows a plan view of the tank and sand pads (see Figures 12a and 12b). DOE Idaho explained that only extreme measures would allow sand pad samples to be collected and that the samples would be highly variable and uncertain, making it difficult to use the sampling data to predict the inventory in the sand pad. NRC staff appreciates the difficulty in sampling the sand pad and concludes DOE Idaho made a good faith effort to evaluate the uncertainty in the sand pad inventory through additional sensitivity and uncertainty analysis, modeling, and analysis of indirect sampling data. The sand pad inventory is adequate for the purpose of assessing DOE Idaho's demonstration of compliance with NDAA Criteria 2 and 3.

3.2.7 Evaluation of Estimated Auxiliary Equipment Inventory

NRC staff has confidence that the auxiliary equipment inventory developed by DOE Idaho for the piping, valve boxes, associated secondary containment, and vaults will not significantly impact or underestimate the overall risks[7] associated with the TFF. Because the potential risk from the sand pad and large tank sources is substantially higher, the focus of this TER is on the sand pads and tanks.

3.3 Identification of Highly Radioactive Radionuclides

DOE defined "highly radioactive radionuclides (HRRs)" to be those radionuclides that, using a risk-informed approach, contribute most significantly to radiological risk to workers, the public, and the environment using sensitivity analysis (DOE Idaho, 2005). To identify HRRs, DOE Idaho started with an initial list of 145 radionuclides and compared it to the list of radionuclides

[7]The piping inventory is important for the intruder construction scenarios, but the expected dose is low and is significantly lower than the intruder driller scenario (see Sections 4.2.14 and 4.2.15).

found in Tables 1 and 2[8] in 10 CFR 61.55[9]. Although not specifically discussed in the waste determination (DOE Idaho, 2005), this would also entail correctly identifying the group of radionuclides that would fit into the category of alpha-emitting, transuranic nuclides with half-lives greater than five years (10 CFR 61.55, Table 1) in DOE Idaho's list of HRRs. Additionally, DOE Idaho did not include short-lived radionuclides with half-lives less than 5 years as HRRs, although this group of radionuclides is also specifically identified in 10 CFR 61.55, Table 2.

DOE Idaho used a screening process to identify additional HRRs that most significantly contribute to radiological risk to the public, including inadvertent intruders. Radionuclides were initially screened based on half-life. All radionuclides and associated decay chain members, if the radionuclide was a member of a chain, were screened from the list of potential highly radioactive radionuclides for public dose if the half-life and half-lives of associated decay chain members (for radionuclides in a decay chain) were all less than 5 years. Radionuclides were also screened from DOE Idaho's list of HRRs if their half-lives were sufficiently long to be considered stable or if the specific activity was so low that the dose contribution would be insignificant (see list of screened radionuclides in Section 5.1.2.1 of DOE Idaho, 2005).

Next, DOE Idaho performed additional screening analyses for the groundwater pathway to identify those radionuclides that contributed most significantly to public dose (see Sections 4.2.12 and 4.2.13 for more information on the public receptor scenario). For the groundwater pathway screening, DOE Idaho calculated the expected pore water concentration in the wasteform (e.g., grouted waste or sand pad) to determine whether this concentration, if consumed as drinking water, would lead to an annual effective dose equivalent of 4 mrem/yr [4×10^{-2} mSv/yr], assuming a consumption rate of 2 L/d [70 oz/d]. Twenty-nine radionuclides remained following this screening. Of the remaining 29 radionuclides, DOE Idaho used DUST-MS and GWSCREEN (Rood, 1998) to determine release rates and resultant groundwater concentrations for these radionuclides. The results of this analysis are presented in Appendix F of the PA (DOE Idaho, 2003a). Only Sr-90, Tc-99, and I-129 contributed significantly to the expected groundwater all-pathways dose (expected to contribute more than 99 percent of the peak dose). Therefore, only these radionuclides were retained for detailed groundwater analysis using the PORFLOW computer code (Runchal, 1997). These radionuclides were already listed by default as HRRs based on their specific inclusion in 10 CFR 61.55, Tables 1 and 2.

DOE Idaho also performed calculations in its PA to determine those radionuclides that contribute most significantly to the expected inadvertent intruder dose. DOE Idaho did not identify any additional HRRs as a result of these calculations. See Table 3 for a list of HRRs.

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[8]Among other waste classification limits (e.g., Class A and Class B), 10 CFR 61.55, Table 1, includes Class C activity concentration limits for long-lived radionuclides, and Table 2 includes Class C activity concentration limits for short-lived radionuclides with the exception of short-lived radionuclides with half-lives less than 5 years, H-3, and Co-60, which can be no higher than Class B by themselves. The NDAA only requires a determination of whether the waste is greater or less than Class C.

[9] DOE noted that although Tables 1 and 2 in 10 CFR 61.55 specify concentration limits for certain radionuclides in the form of activated metal (e.g., C-14, Ni-59, Nb-94, Ni-63), DOE included these radionuclides regardless of whether the radionuclide was in the form of activated metal (DOE Idaho, 2005).

Table 3. List of Highly Radioactive Radionuclides (Modified From Table 5, DOE Idaho, 2005)			
Radionuclide	Radionuclide Half-Life (yr)	Long-Term Radiation Hazards	Short-Term Radiation Hazards
Am-24*†	4.3×10^{2}	X	
C-14*†	5.7×10^{3}	X	
Cm-242†‡	4.5×10^{-1}	X	
Co-60§	5.3		X
Cs-137*§	3.0×10^{1}		X
Ba-137m*	4.9×10^{-6}		X
H-3*§	1.2×10^{1}		X
I-129*†	1.6×10^{7}	X	
Nb-94†	2.0×10^{4}	X	
Ni-59†	7.5×10^{4}	X	
Ni-63†	1.0×10^{2}		X
Np-237*†	2.1×10^{6}	X	
Pu-238*†	8.8×10^{1}	X	
Pu-239*†	2.4×10^{4}	X	
Pu-240*†	6.6×10^{3}	X	
Pu-241†‡	1.4×10^{1}	X	
Pu-242†	3.8×10^{5}	X	
Sr-90*§	2.9×10^{1}		X
Y-90*	7.3×10^{-3}		X
Tc-99*†	2.1×10^{5}	X	

*HRRs based on dose assessment results from the 2003 PA (DOE Idaho, 2003a).
†Taken from Table 1 of 10 CFR 61.55.
‡Daughter product of long-lived radionuclide.
§Taken from Table 2 of CFR 61.55.

3.4 NRC Review and Conclusions–Identification of HRRs

The NRC staff reviewed DOE Idaho's approach for identifying HRRs and determined that it is generally consistent with NRC's recommendations for identification of HRRs as presented in NRC's draft guidance, "Standard Review Plan for Activities Related to U.S. Department of Energy Waste Determinations" (NRC, 2006b). The NRC staff does have some concerns regarding DOE Idaho's implementation of this approach, however, as discussed next.

Based on NRC staff's review of the list of radionuclides in Table A-12 of the Draft Section 3116 Determination (DOE Idaho, 2005), NRC identified one alpha-emitting, transuranic radionuclide with a half-life greater than 5 years that was present in significant enough activity {5.9×10^{2} MBq [1.6×10^{-2} Ci]}, to potentially pose a risk to human health or to affect the waste classification. Although Cm-244 was present at a high enough activity to remain in the list of radionuclides after the initial screening, Figure F-58 in the PA (DOE Idaho, 2003a) presents the results of the additional GWSCREEN modeling which indicate negligible concentrations of Cm-244 are expected in groundwater. The predicated low Cm-244 groundwater concentrations are due to high sorption coefficients of Cm-244 in grout, and perhaps in the vadose zone materials, although vadose zone sorption coefficients were not provided. Although Cm-244 was included in the inadvertent intruder analysis, it was not found to be a significant dose contributor (DOE Idaho, 2003a). Cm-244 was evaluated by NRC in its waste classification calculations and is discussed further in Section 4.1.

Although DOE Idaho did not list any short-lived radionuclides with half-lives less than 5 years (a category of radionuclides listed in 10 CFR 61.55, Table 2), NRC staff agrees with DOE Idaho's screening methodology based on half-life. The acceptability of this screening approach is based on the expectation that controls will be in place to limit exposures to workers and members of the public during the 100-year institutional control period after closure when short-lived radionuclides are potentially present in significant enough quantities to pose a human health risk. Thus, NRC staff concludes that DOE Idaho included all radionuclides in its list of HRRs that are specifically identified in 10 CFR 61.55, Tables 1 and 2, and those alpha-emitting transuranic radionuclides with half-lives greater than 5 years that are important to radiological risk to the public (including inadvertent intruders), workers, and the environment.

It is important to note that NRC is not endorsing the sole use of 10 CFR 61.55, Tables 1 and 2, as the most appropriate method of identifying HRRs. However, NRC staff recognizes that the dual use of the HRR list, both to identify those radionuclides that must be removed to the maximum extent practical and to determine those radionuclides to be targeted for sampling for waste classification purposes, makes this approach appealing, as it may help facilitate demonstration of compliance with NDAA criteria. On the other hand, inclusion of all the radionuclides listed in 10 CFR 61.55, Table 1 and 2, may actually focus attention on radionuclides that may not be risk-drivers with respect to meeting the performance objectives in 10 CFR 61, Subpart C. It is significant to note that all of the HRRs identified by DOE Idaho, including those radionuclides found to be most significant to public and inadvertent intruder doses based on DOE's PA results, are listed in 10 CFR 61.55 with two exceptions. These exceptions are two radionuclides that are daughter products of radionuclides listed in 10 CFR 61.55, Table 2 (i.e., Y-90 and Ba-137m, daughter products of Sr-90 and Cs-137, respectively). Several HRRs were listed simply because they were found in 10 CFR 61.55, Tables 1 and 2, although they were not expected to contribute significantly to the dose based on the performance assessment results (e.g., Cm-242, Co-60, Ni-59, Ni-63, Pu-241, and Pu-242).

DOE Idaho conservatively used the worst-case tank inventory in the PA uncertainty analysis for the screening analysis (DOE Idaho, 2003a). The screening analysis for the sand pad used the conservative or compliance case one-dimensional diffusion modeling presented in the PA (DOE Idaho, 2003a). However, DOE Idaho recently performed additional sensitivity modeling for the sand pad inventory in response to NRC's RAIs 3 and 4 (NRC, 2006a). Based on the results of this modeling, some radionuclide inventories could be significantly (order of magnitude) higher than the base case sand pad inventory assumed in the PA (DOE Idaho, 2003a). Of these radionuclides, Np-237 had a high enough activity in the sand pad to be a potential HRR for the groundwater pathway. NRC staff requested additional clarifying information (NRC, 2006c) regarding the Np-237 screening analysis at the June 1, 2006, meeting to discuss DOE Idaho's RAI responses (DOE Idaho, 2006a). In response to NRC staff's additional information request (NRC 2006c), DOE Idaho provided information regarding DUST-MS release calculations that showed the chemical barrier afforded by the concrete vault floor significantly delayed the release of parent radionuclides, Pu-241 and Am-241, in the decay chain, as well as Np-237 (DOE Idaho, 2006c). Additionally, long-term groundwater concentrations for Pu-241, Am-241, and Np-237 estimated with GWSCREEN showed that the peak groundwater concentrations for these constituents was not expected to occur until after the period of performance of 10,000 years (Appendix F, Figure F-62 of the PA, DOE Idaho, 2003a). Although uncertainty in transport parameters was not considered in the screening analysis, NRC staff attempted to reduce the uncertainty in the transport of Pu-241, Am-241, and Np-237 as discussed below. Furthermore, Pu-241, Am-241, and Np-237 were already included as HRRs by default, as they are included in Tables 1 and 2 in 10 CFR 61.55, although they were not specifically targeted for detailed groundwater analysis.

The list of HRRs developed by DOE Idaho for the groundwater all-pathways dose did not consider the uncertainty of key transport parameters in the screening process. The NRC staff was concerned that if DOE Idaho had performed sensitivity or uncertainty analyses on transport parameters during the screening process, DOE may have identified additional HRRs important to meeting the 10 CFR 61.41 performance objective. To address this concern, NRC staff performed its own independent assessment. The assessment included review of recent groundwater monitoring time series data, and analysis of existing contamination from historical SBW releases that resulted in direct contamination of the subsurface underneath the TFF. The NRC's staff's analysis of this data shows that Sr-90, Tc-99, I-129, and C-14 are the primary contaminants currently detected at elevated concentrations in perched groundwater underneath the TFF (DOE Idaho, 2006d; DOE Idaho, 2006e). Recent characterization showed an estimated concentration of 0.18 pCi/L of Np-237 in just one well downgradient of the TFF. This elevated concentration was most likely a result of historical service water injection well operations. Concentrations of radionuclides from historical releases are generally expected to be higher than future releases of contaminants in the grouted wasteforms, due primarily to the chemical barrier afforded by the grout and concrete (see Sections 4.2.6 and 4.2.7 for additional discussion). Therefore, the NRC staff addressed the uncertainty in the list of HRRs for the groundwater pathway with monitoring data from historical releases of primarily SBW from the TFF.

Finally, HRRs for worker doses were not specifically identified in the Draft Section 3116 Determination (DOE Idaho, 2005). In response to clarifying RAI 18 (NRC, 2005), which specifically asked DOE Idaho to list key (or highly radioactive) radionuclides for worker dose and clarify whether short-lived radionuclides were screened out in the identification process, DOE Idaho stated that the screening methodology used in the PA (DOE Idaho, 2003a) was for post-closure public (including inadvertent intruder) doses and not for worker evaluations. However, DOE Idaho did not communicate which worker evaluations had been performed, if any, or any screening for these evaluations. DOE Idaho did state that worker doses would occur from removal of existing equipment, installation of cleaning equipment, and sampling. DOE Idaho presented information regarding worker doses from previous cleaning operations (see Section 4.2.16 for additional details) and stated that worker doses would be primarily from high-energy gamma emitters such as Cs-137. Worker doses are also minimized and maintained as low as is reasonably achievable (ALARA) through use of remote cleaning operations. In the June 1, 2006, meeting, NRC staff asked DOE Idaho to specifically identify highly radioactive radionuclides for worker dose. DOE Idaho responded (NRC, 2006d) that the HRRs for worker dose are those radionuclides found in Table 5 of the Draft Section 3116 Determination (DOE Idaho, 2005). NRC staff agrees that HRRs for worker doses are expected to be short-lived, high energy gamma emitters such as Cs-137 (gamma from daughter product Ba-137m). NRC staff also thinks that radionuclides that pose an inhalation risk are potential HRRs for worker dose (e.g., Pu-238, Pu-239, Pu-240, and Am-241). As a result of the NRC staff's review and the DOE Idaho's responses to this matter, NRC staff concludes that all HRRs (Table 3) have been identified.

3.5 Alternative Waste Treatment Technologies

The NRC staff evaluated the technology selection process in 2003 (NRC, 2003b), prior to initiation of cleaning operations. In making the technological selections for the INTEC TFF, DOE Idaho previously evaluated numerous chemical and mechanical waste removal technologies for cleaning of HLW tanks, primarily as part of its participation in the TFA Technical Team. This national group was developed to assess tank cleaning technology throughout the DOE complex. As a result of the basic research on tank cleaning technologies evaluated complex-wide, DOE Idaho did not need to conduct significant additional basic research on tank cleaning at the time.

The TFA technical team coordinated with all DOE sites to develop cleaning processes based on site needs. The equipment developed had to fit inside the tanks, be compatible with the tank environment, and be able to clean the specific tank waste. Tank cleaning could be accomplished using either chemical or mechanical processes.

Chemical processes the TFA Team considered included the following (DOE Idaho, 2002d):

- Caustic Recycle—An electrolytic process that selectively separates sodium ions from a waste stream to reduce the overall quantity of waste that must be treated for disposal.

- Sludge Washing—A chemical process for washing with Fenton's Reagent (a mixture of hydrogen peroxide with an iron catalyst) that destroys ion-exchange resin to release waste absorbed on the resin and allow it to be treated for disposal.

- Saltcake Dissolution—A process for dissolving crust level growth in the Hanford SY-101 tank.

- Chemical Cleaning—A process using various organic acids, possibly combined with caustic leaching, to remove aluminum compounds and dissolve portions of dense heel solids. By breaking up the solid mass, the resulting slurry can then be pumped out of the tank.

- Enhanced Sludge Washing—A chemical process that involves a series of washes where tank waste is mixed with aqueous solutions containing sodium hydroxide. The waste solution is heated and cooled. Then, the liquid containing the nonradioactive elements is decanted.

DOE Idaho determined that chemical treatment processes were not practical for cleaning the TFF tanks. Caustic recycle, solids washing, and saltcake dissolution were developed for the neutralized waste at other DOE sites and did not apply to the TFF acidic waste, because volume reduction is not a concern for removing HRRs to the maximum extent practical in TFF waste, ion exchange resin is not present in TFF tanks, and waste does not adhere strongly to TFF tank surfaces. An acid strong enough to dissolve the existing solids could also cause tank corrosion. Washing the solids in a basic solution could cause more precipitates, which would further aggravate the solids problem. Based on this evaluation, DOE Idaho concluded that chemical treatment processes were not appropriate for the TFF.

Mechanical processes the TFA Team considered included

- Mixer Pumps—High-pressure pumps that intake and discharge sludge in the tank bottoms to slurry the mixture and allow it to be pumped from the tank. Various systems were developed and tested.

- Sluicing Systems—High-pressure water systems that slurry the sludge and move it toward discharge pumps. Various types of sluicing systems were considered.

- Disposable Crawler—Commercially developed motorized treads that break up and mobilize the sludge. A sluicer mounted on top of the motorized treads then uses a high-pressure water jet to move the loosened material toward a transfer pump.

- Mechanical Arms—Robotic arms installed through tank risers that are capable of deploying in-tank surveillance, confined sluicing, inspection, and waste analysis tools called end effectors.

The solids at INTEC are well dispersed in the residual liquid materials and have not been observed to adhere strongly to tank surfaces. The disposable crawler was determined not to be technically practical because of interference of cooling coils located on the bottoms of most of the tanks. Most pumping systems were developed to remove large quantities, rather than small quantities of liquids with small amounts of suspended solids, and thus were not technically practical for INTEC waste. Because these technologies were determined not to be technically practical, they were not retained for evaluation of economic practicality.

Sluicing systems and mechanical arms were determined to be technically practical and were retained for economic evaluation. Specifically, a high-pressure water system consisting of a spray ball (washball), directional spray nozzles, and a mechanical arm that could hold a video camera for in-tank inspections and equipment for sampling, combined with a modified[10] steam-jet pumping system, was determined to be a practical alternative for TFF waste. The washball was developed commercially by the chemical and oil industries for tank cleaning. The TFA introduced it to the DOE complex, and DOE Idaho has deployed it successfully for cleaning the TFF tanks. The INL Site borrowed directional nozzle technology from the TFA and other sites to develop a remotely operable sluicing system that consists of a high-pressure spray nozzle, lights, and video camera that can be extended into a tank on a long pipe (PNNL, 2001).

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[10]The steam jet was modified by cutting the steam-supply line and installing a new steam jet lower in the tanks.

Prior to deployment, the INL tank cleaning system was tested in a full-scale mockup tank using simulated waste (DOE Idaho, 2001). The testing showed the effectiveness of the remote camera, lights, and washball to remove surrogate solids from the tanks. While very effective at removing most of the surrogate waste from the tank, the washball had a tendency to push the waste out from the center of the tank towards the tank walls. The cooling coils also inhibited the water and solids from moving or settling back toward the center of the tank. Thus, the directional nozzles were useful at removing additional solids near the tank wall. The mock demonstration gave DOE Idaho the opportunity to test various nozzle sizes and determine the best operational parameters (e.g., optimal flow ring vane configuration, nozzle size, and pressure and flow rate ranges). DOE Idaho deployed the system to clean tanks WM–180 through WM–186 from 2002 to early 2005 (DOE Idaho, 2005). The results of these cleaning activities are discussed further in Section 3.7 and show that the system performed much better than assumed for the conservative case in the DOE PA (DOE Idaho, 2003a) to demonstrate compliance with the performance objectives (DOE Idaho, 2005).

In response to NRC's RAI 5, DOE Idaho indicated that the project continues to review technology used in the DOE complex and available new technology through participation in technical exchanges and weekly conference calls with the DOE Environmental Management Office. DOE Idaho has not identified a new technology that has not previously been considered (DOE Idaho, 2006a).

3.6 NRC Review and Conclusions–Alternative Waste Treatment Technologies

The technical practicality of waste removal options focused on mechanical and chemical processes. Emphasis was placed on the specific chemical and physical form of the INTEC wastes when evaluating the available technologies. Because the INTEC waste is acidic, it would not likely be technically practical to pursue bulk chemical cleaning. Analysis of cleaning activities conducted from 2002 to early 2005 show that the mechanical processes selected for bulk waste removal have been very effective (see Figure 14). The tank cleaning technology DOE Idaho selected appears to be a good choice for removing HRRs to the maximum extent practical, as indicated by the removal estimates provided in Section 3.7.

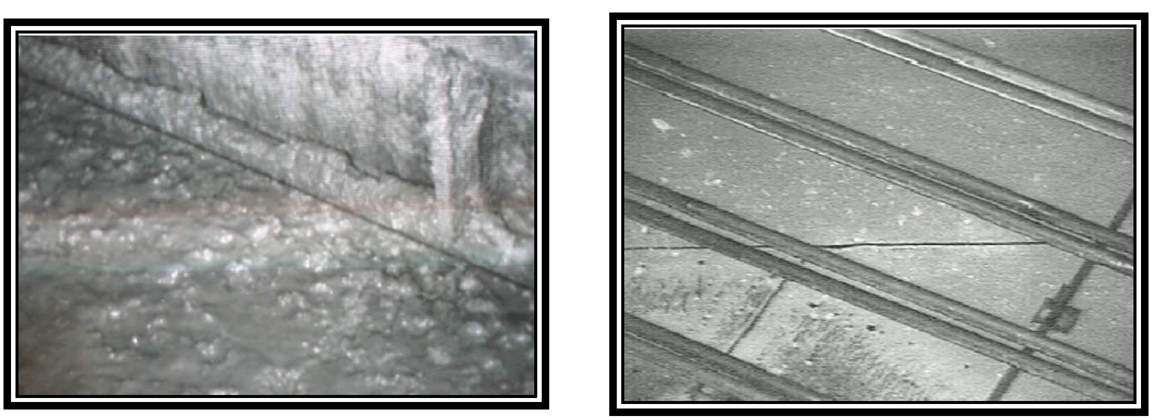

Figure 14. Image of Tank Surfaces Before Cleaning and After Cleaning (From DOE Idaho, 2005)

3.7 Removal to the Maximum Extent Practical

Prior to cleaning and subsequent sampling, DOE Idaho used planning documents such as the first HWMA/RCRA closure plan for WM–182 (DOE Idaho, 2003b) and the TFF PA (DOE Idaho, 2003a) to establish a baseline inventory and "goals" for the cleaning activities to meet or exceed. DOE Idaho contends that it attempted to remove as much activity from the tanks as possible and thus did not establish "goals" such as volume, mass, or activity limits for remaining residuals in the cleaned tanks. The planning documents established a baseline inventory that DOE knew would meet performance objectives. These planning documents and mock-up cleaning demonstrations provided DOE Idaho with confidence prior to full-scale deployment of the cleaning technology that the final end-state of tanks at the TFF after cleaning could meet applicable criteria. However, because compliance with performance objectives does not necessarily obviate the need for additional cleaning, it would have been inappropriate for DOE Idaho to establish goals that met the performance objectives with no further analysis of the practicality of additional removal.

3.7.1 Cleaning Process and Criteria for Termination of Cleaning Operations

In response to NRC's RAI 5 regarding the basis for the determination that HRRs had been removed to the maximum extent practical and that cleaning operations could be terminated, DOE Idaho provided additional information on how it determined that it had exhausted the capability of the selected technology to remove significant quantities of additional activity from the tanks (DOE Idaho, 2006a). This information serves as one line of evidence for the practicality of additional removal with respect to the capability of the selected technology. During cleaning activities, cameras and video recording are used to help identify areas targeted for cleaning. DOE Idaho explained that operators and supervisors review the videos of cleaning and view real-time cleaning on cameras to maximize performance during the cleaning cycle. In addition, radiation levels are monitored with an unshielded Geiger-Mueller counter near the pipe on the steam-jet transport line to determine the relative effectiveness of system operations. Each day, the tank was visually inspected to map the areas that contained the largest volumes to target for additional solids removal. Project personnel met to review the remote video inspection and used the directional nozzles to tackle those areas.

Initially, only the washball is used to remove solids. However, after several thousand gallons of cleaning water is introduced, the washball becomes far less effective (see Figure 10 curve for WM–182). Two (or sometimes three) directional nozzles are then substituted for the washball. A spike in the effluent activity per gallon of wash water removed occurs immediately as the solids removal efficiency increases significantly with introduction of the directional nozzles (see Figure 10). After tens of thousands of gallons of water are flushed through the system, essentially no measurable level of radioactivity is removed with the wash water. When radiation levels decrease to a low, constant value, cleaning is terminated, and the tanks are visually examined using remote video inspection. In practice, even after the radiation levels decrease to their lowest level and remain constant, tank washing continues for another day by flushing with several thousand gallons of water (DOE Idaho, 2005).

DOE Idaho accumulated lessons learned during the mockup cleaning demonstration, the demonstration of the technology with tank WM–182, and throughout the entire cleaning process from 2002 through early 2005. For example, based on observations made during tank cleaning activities performed to date, DOE Idaho concludes that directional nozzles are most effective at agitating tank contents to facilitate solid removal when directional nozzles are used with approximately 19-27 m^3 [5,000-7,000 gal] of water in the tank for a short time (100–140 minutes). This technique ensures that solid particles are suspended in the liquid for a period comparable to the time required to remove the liquid and suspended solids by pumping (DOE Idaho, 2000).

3.7.2 Explanation of Differences in Cleaning Effectiveness for Individual Tanks

The NRC staff also requested information (NRC, 2006a) regarding the various patterns of residual contamination reflected in the EDFs for each cleaned tank (see DOE Idaho, 2005, Section 2.4.2, for a list of references). DOE Idaho was also asked to discuss limitations in geometry, such as cooling coils, that made it more or less difficult to clean individual tanks. DOE Idaho explained that the configuration of the tank bottoms vary for each tank. Eight of the eleven 1,000-m^3 [300,000-gal] tanks that contain cooling coils are similar in design, but differences exist in the exact location of cooling coil routing and the number of cooling coil supports. These differences present slightly different challenges in terms of the ability to move tank solids around the tank floor. DOE Idaho stated that these obstructions explain the higher amount of solid residuals in WM–182 (DOE Idaho, 2006a). On the other hand, three of the 1,000-m^3 [300,000-gal] tanks do not contain cooling coils and presented fewer challenges in removing tank solids through the steam jets. The remote video inspection of the tank bottoms show areas that are slightly lower than others, creating a wave or rippling effect which results in the tank having low spots in differing locations on the tank floor. For example, remote visual inspection shows that the outer edge of the floor of tank WM–185 is several inches lower than the center of the tank, which resulted in solids settling around the outer edge of the tank floor (see Figure 13). When the tank residual waste was emptied to its lowest level, bare spots were observed in the center of the tank floor. DOE Idaho noted that there were no areas of the tanks where the wash water could not move solids. As the cleaning was terminated in each tank, several inches of flush water was left in the tanks, and any solids that could not be pumped out of the tank settled onto the tank floor. These solids settled out differently in the various tanks depending upon where the water from the spray nozzles and washball last contacted the tank floor and where the low spots and interferences exist on the tank bottom. Tank bottom areas that have more solids are usually the low spots in the tank, and the cleanest areas usually are the last spot the nozzle sprayed or a high spot on the floor.

In summary, differences in the kriging maps for each tank demonstrate slight differences in cleaning effectiveness between tanks, as well as differences in the exact operation of the spray wash components. Lastly, differences in the final configuration of remaining tank solids in the various tanks are due to incorporation of lessons learned, which resulted in some changes in the operational parameters during cleaning. As the cleaning activities progressed to different tanks, lessons learned meetings were held to discuss changes that could be used to increase cleaning efficiency and productivity. The major lesson learned from the first tank cleaning efforts was to remove as much residual waste as possible with the new steam jet before cleaning started. Additionally, cleaning of the first two tanks required slow removal of tank heels and solids to allow for evaluation of radiation levels in the transfer piping and valve boxes. After cleaning these first two tanks, it was determined that the radiation levels would not impact operational activities, which allowed for removal of heels as rapidly as possible. This new approach did not result in a cleaner tank but required less water to complete the cleaning and explains why effluent concentration curves have much smaller cumulative volumes at the end of cleaning operations (see Figures RAI 5–A–1 through RAI 5–A–7, DOE Idaho, 2006a).

In short, DOE Idaho attempted to optimize system operation over time to ensure removal to the maximum extent practical. The success of cleaning activities is reflected in the fact that there is no evidence of any buildup of residual on the side walls of the tanks after cleaning, and large areas of the tank floors are bare (see Figure 14). Only a fine layer of residual solid, approximately 0.97 cm [0.38 in], appears on some areas of the cleaned tanks. DOE Idaho stated that additional spray cleaning was not able to remove the small quantity of residual remaining in the tanks (DOE Idaho, 2006a).

DOE Idaho provided a table (see Table 4) that showed the percentage of removal at closure (2012) of all HRRs identified for the TFF. It is important to note that the cleaning technology did not target particular radionuclides and only attempted to achieve bulk mass removal. However, Sr-90 concentrations decreased significantly in the solid residual heels due to the apparent preferential removal of Sr-90 by dissolution in the large quantity of flush water used to clean the tanks. Removal efficiencies exceeded 99 percent for most HRRs. It is also significant to note that DOE Idaho was not able to provide an estimate of the actual removal efficiency following deployment of the washball and directional nozzle technology due to lack of development of an inventory prior to cleaning operations for each tank. Therefore, the actual effectiveness of the cleaning technology is uncertain. DOE Idaho estimates that approximately 90 percent of the tank solids were removed and the efficiency in removal of highly soluble radionuclides such as Sr-90 is expected to be significantly higher (DOE Idaho, 2006a). Almost the entire remaining inventory in the tanks is in the solid heels, and liquid heel concentrations were substantially diluted in the flush water. Precipitation of SBW due to the introduction of large quantities of demineralized flush water was evaluated but not expected to be significant (Millet, et al., 2005). The concerns (expressed in Section 3.2) that Cs-137 concentrations in tank WM–188 are expected to be much higher than the concentration of Cs-137 in tank WM-182 used to estimate the inventory for the uncleaned tanks and that tank WM–187 may contain a much larger quantity of solid residual heel to be removed are mitigated by the likely overestimation of the inventory of a relatively clean tank used as a spare (tank WM–190). Furthermore, DOE Idaho may have been overly conservative in assigning activity to the interstitial liquid portion of the solid volume based on the solid sampling, as discussed in Section 4.1.

3.7.3 Costs—Worker Dose

DOE Idaho provided information regarding one cost of radionuclide removal, i.e., worker dose. Worker doses associated with closure activities for tank WM–182 include equipment removal and installation, cleaning operations, sampling, and grouting. DOE Idaho estimated exposures from cleaning a TFF tank based on radiation exposure information on tank cleaning operations from Jacobson (2002) and a review of radiation work permit electronic dosimetry results for January 1, 2002, through June 15, 2005 (Martin, 2005). Worker doses for WM–182 totaled 6.11 mSv [611 mrem] [this also includes 0.15 mSv [15 mrem] for removing equipment and preparing tank WM–183 for a washball test]. A total of 49.3 person-mSv [4,931 person-mrem] were recorded for TFF work performed from January 1, 2002, through June 15, 2005 for all tanks. Maintenance activities for auxiliary equipment account for 25.68 person-mSv [2,568 person-mrem] or about 52 percent of the total worker exposure. If the entire balance of 23.63 person-mSv [2,363 person-mrem] is attributed to tank cleaning for the seven tanks, exposures average 3.38 person-mSv/tank [338 person-mrem/tank]. The actual exposure of

Table 4. Percentage of HRRs Removed From All Tanks and Ancillary Equipment (modified from Table 6, DOE Idaho, 2005)			
Radionuclide	**Total Ci Generated at INTEC**	**Residual Ci* in Tanks at Closure†**	**Percent Removed at Closure‡**
Am-241§	9.28×10^{3}	6.97	99.92%
Ba-137m§	8.95×10^{6}	1.19×10^{4}	99.87%
C-14	2.91×10^{-2}	3.85×10^{-5}	99.87%
Cm242	1.51×10^{1}	1.00×10^{-2}	99.93%
Co-60	1.67×10^{3}	4.79×10^{-1}	99.97%
Cs-137§	9.46×10^{6}	1.19×10^{4}	99.87%
I-129§	6.01	5.87×10^{-3}	99.90%
H-3	7.13×10^{3}	5.43	99.92%
Nb-94	1.54×10^{3}	1.60	99.90%
Ni-59	3.71×10^{3}	1.90×10^{-1}	99.99%
Ni-63	4.36×10^{5}	2.17×10^{1}	99.99%
Np-237§	7.53×10^{1}	3.57×10^{-1}	99.53%
Pu-238§	1.07×10^{5}	9.08×10^{1}	99.92%
Pu-239§	2.83×10^{3}	2.90×10^{1}	98.98%
Pu-240	1.46×10^{3}	1.09×10^{1}	99.25%
Pu-241	4.73×10^{4}	1.52×10^{2}	99.68%
Pu-242	3.94	7.60×10^{-3}	99.81%
Sr-90§	8.42×10^{6}	6.78×10^{2}	99.99%
Tc-99§	3.67×10^{3}	5.79	99.84%
Y-90§	8.42×10^{6}	6.75×10^{2}	99.99%
Total (Ci)‖	3.59×10^{7}	2.48×10^{4}	99.93%

*1 Ci = 3.7×10^{4} MBq.
†Total Ci at closure includes Ba-137m and Y-90 and radionuclide decay to 2012 based on: (i) heel residuals that are estimated using remote video inspection of cleaned tank internals to map out estimates of depth of remaining residual solids and liquids across tank bottoms using tank internal reference points of known height, (ii) best estimated radionuclide concentrations from past and recent samples, and (iii) radioactive decay to 2012.
‡The removal efficiencies are based on a baseline inventory that included all waste generated at INTEC and stored in the TFF.
§Radionuclides that are significant contributors to dose calculations in the 2003 TFF PA (DOE Idaho, 2003a).
‖Radionuclides shown are contributors to the dose calculations or regulated by concentration limits in 10 CFR 61.55. The totals are based on the entire inventory of radionuclides.

6.11 person-mSv [611 person-mrem] from cleaning the most contaminated tank (WM–182) is a reasonable dose projection for future tank cleaning (Martin, 2005). Twenty-three personnel involved directly with tank cleaning received a radiation dose from TFF closure activities. The maximum exposure for any worker to date is 1.17 mSv [117 mrem].

Based on the information above, the following is concluded:

- The average radiation exposure that will be experienced for cleaning and closing each TFF tank is expected to total about 6.5 mSv [650 mrem] for all occupational exposure.

- The exposure per person for cleaning a TFF tank will be about 6.5 mSv [650 mrem] divided by 23 people, which is about 0.3 mSv [30 mrem] per person.

- Maximum radiation exposure for an individual worker is estimated to be 1.2 mSv [120 mrem] for cleaning a single TFF tank.

- Worker dose for tank cleaning is minimal because all cleaning is accomplished remotely.

- A total exposure of about 7.15×10^{-2} Sv [7.15 rem] is expected for cleaning eleven large tanks.

Worker exposure for complete tank removal would occur from excavating the tanks, cutting them up, and packaging them for disposal. Worker exposure for complete removal of the TFF tank system is estimated to be 10.7 mSv/yr/worker [1,070 mrem/yr/worker] for an average of 326 workers/year for an estimated 26 years for a total exposure of over 90 Sv [9,000 rem] (DOE Idaho, 1998b).

It is assumed that development and deployment of a new technology to remove additional radionuclides would lead to higher worker doses, due to potential soil excavation, equipment removal or installation, additional cleaning, sampling, etc. On the other hand, the projected dose to the public (including inadvertent intruders), which is already expected to be low, may only be lowered slightly (see following discussion).

3.7.4 Costs—Technology Development and Deployment

DOE Idaho provided a cost estimate of $46.1 million (non-escalated, see Table 5) for development of a new technology in an effort to analyze the practicality of attempting additional waste residual removal from cleaned tanks. This figure is based on a 1998 report that estimated the costs of various options for closure of the TFF (DOE Idaho, 1998b). The actual expenditure for TFF closure activities, which included development of the cleaning technology and actual cleaning of seven large tanks and four smaller tanks from 1999 to 2005, is only $35 million. It is expected to cost an additional $1 million for each of the four uncleaned tanks (DOE Idaho, 2005).

DOE Idaho expects development and deployment of a new technology to cost slightly more than development and deployment of the existing tank cleaning system to date. The estimated cost was based on a 10-percent escalation rate of the cost of the current technology development and deployment, or $38.5 million, but the costs associated with a new technology might be significantly higher. The current tank cleaning system that has been deployed at the

Table 5. Estimated Cost of TFF Technology (Modified From Table 7, DOE Idaho, 2005)*	
Activity	**Estimated Cost (Millions of Dollars)**
Design of System	19.0
Proof of Process	5.8
Site Preparation	9.7
Characterization of Waste	4.6
Tank Isolation	7.0
Total	46.1

*The actual cost for cleaning seven large tanks to date is $35 million. The four large, uncleaned tanks are expected to cost an additional $1 million/tank or $4 million total for four large tanks. Thus, the total expected cost of cleaning all eleven large tanks is $39 million.

TFF has used much of the existing equipment in the TFF. A new system may not use all of the existing equipment. Access to tanks through existing risers is a limiting factor for deployment of a new technology. If new access must be designed that penetrates the soils, vault roof, and tank, the cost will increase considerably. Additional worker exposure from disturbed soils and tank contents when installing new risers and associated contamination control activities would add to the costs.

The cost of complete removal is estimated to be several billion dollars (DOE Idaho, 1998b). Complete tank removal would also cause more exposure and result in radioactive waste for which disposal would be difficult. Deployment of alternative tank cleaning technologies or complete removal may also delay closure of the TFF past 2012, leading to additional maintenance costs.

3.7.5 Costs—Additional Waste Generation

DOE Idaho stated that monetary costs of additional cleaning were not a factor in decisions to terminate cleaning operations. However, minimizing the generation of new waste volumes is one of the goals of any DOE activity; therefore, it is not prudent to continue to clean tanks and other TFF components once cleaning becomes ineffective. The largest components of cleaning costs are associated with removal of existing equipment, installation of cleaning equipment, and sampling and analysis activities not with flushing operations. Flush water and removed waste materials from cleaning activities, are jetted to operational waste systems and then to facility evaporator systems for volume reduction. The incremental costs of evaporating the flush water are not significant. Therefore, additional spray cleaning of tank components would not have been prohibitive in terms of cost, but rather, were judged to be ineffective in removal of additional waste.

3.7.6 Benefits of Additional Removal

The most obvious benefit of additional radionuclide removal is risk reduction to members of the public. However, DOE Idaho notes that for those tanks with sand pads, the benefit of additional radionuclide removal is limited by the fact that approximately 43 percent of the inventory is expected to be in the sand pads, which cannot be easily treated. The sand pads may contain most of the Cs-137 and Sr-90 activity, and development and deployment of a new tank cleaning system could only decrease the total inventory by about 60 percent. DOE Idaho presented a table showing the costs and benefits for a range (up to a 60-percent reduction in the total inventory) of removal (DOE Idaho, 2005). The peak groundwater all-pathways dose was assumed to occur over a 50-year exposure period. Based on this information, DOE Idaho estimated that it would cost roughly $2.8 million/mrem reduction. DOE Idaho also stated that with typical average doses to the public from natural sources and medical treatment in the range of 3-4 mSv/yr [300–400 mrem/yr], it is impractical to reduce the estimated dose from the TFF by such a small amount (DOE Idaho, 2005).

3.7.7 Summary of Costs and Benefits

Because only limited technologies are applicable to the INTEC TFF, the economic evaluation presented below only considers three main options: the current system (as described above); a hypothetical new system that could completely remove or stabilize all tank waste; and complete tank removal. Table 6 provides a comparison of the performance objectives and costs and benefits of the options considered. Because DOE Idaho is employing what it considers to be the best available technology, detailed calculations for less effective technologies are not considered. As stated above, the preferred system is a washball, directional nozzles, mechanical arms, and steam jets. DOE Idaho used the cost of development and deployment (expected to be approximately $40 million) of the preferred technology as the estimate for how much it may cost to develop a new technology. The new technology and the economic impact associated with its development are only estimates for the purpose of evaluating costs and benefits. DOE Idaho stated that development of a new technology may not be practical because: (i) the new technology is not yet developed and would most likely delay the 2012 closure date and lead to the incursion of additional maintenance costs, (ii) the performance objectives can be achieved with conservative assumptions in key models or parameters, and (iii) removal efficiencies for HRRs are high for the preferred system. Complete tank removal has a very large economic impact, as well as a large radiological impact to workers. Although worker doses for complete tank removal meet the performance objectives, many more workers would be exposed at a higher rate for a much longer period of time {91 person-Sv [9,100 person-rem]} compared to implementation of the preferred technology {7.2×10^{-2} person-Sv [7.2 person-rem]}.

3.8 NRC Review and Conclusions—Removal to the Maximum Extent Practical

3.8.1 Evaluation of Process to Terminate Cleaning Operations

NRC staff concludes that DOE Idaho has used an acceptable approach to determining when it has exhausted its ability to clean the tanks using the current technology. DOE Idaho continues to learn from past experience to enhance the effectiveness of its selected technology. For example, DOE Idaho used lessons learned from the mockup cleaning to develop the system comprised of a washball and multiple directional nozzles to target problem areas in the tank. DOE Idaho experimented to determine the optimal depth of flush water in terms of keeping enough depth to maintain solid particles in a slurry for removal, since too much flush water would prevent effective movement of solids by the system. Cameras installed on the spray cleaning equipment allowed real-time visual examination of the tank internals. Tank cleaning was to continue until (i) radiation levels decreased to the lowest value and remained constant, which would indicate that further flushing of the tank internals provided no further waste removal (see Figure 10); and (ii) comprehensive remote visual examinations of the tank internals after each day's cleaning showed that the spray washing was no longer removing further waste residuals (DOE Idaho, 2006a). While use of a radiation detector is expected to be a valuable tool, the efficiency of the instrument is unknown and is expected to be low (the GM detector is located on the outside of the piping and thus, would only be able to detect high energy gamma emitters). Consequently, the use of visual tools to ensure that all areas where significant solid residuals remain are targeted for cleaning is an important part of the demonstration that HRRs have been or will be removed to the maximum extent practical at INL. Additionally, optimization of the operational parameters for the cleaning system provides additional confidence that HRRs have been removed to the maximum extent practical.

3.8.2 Evaluation of HRR Removal Efficiencies

Because no estimates of the inventory were developed prior to cleaning operations, only an uncertain estimate can be presented for HRR removal efficiency related to washball and directional nozzle operation. DOE Idaho estimates that approximately 90 percent of the solid residual volume was removed during cleaning operations, and the efficiency in removal of highly soluble radionuclides such as Sr-90 is expected to be significantly higher due to the dissolution of Sr-90 in the flush water (DOE Idaho, 2006a). This figure is fairly consistent with the estimates from the WM–182 cleaning demonstration (Kimmett, 2002) which predicted a removal of approximately 88 percent of the bulk solids. DOE Idaho can reduce the residual liquid heels relatively easily by lowering the suction level of jet pumps to 3 cm [1 in] from the bottom of the tanks. Because almost the entire remaining inventory is in the residual solid heels, the removal efficiency is estimated as 90 percent or higher for more soluble radionuclides. The concern expressed in Section 3.2 about using tank WM–182 Cs-137 concentrations to estimate the inventory in uncleaned tanks, particularly WM–188, is mitigated by using the same assumption for a relatively clean tank that has been used as a spare during TFF operations (tank WM–190). Furthermore, DOE Idaho may have been overly conservative in assigning activity to the interstitial liquid portion of the solid volume (see Section 4.1).

3.8.3 Evaluation of Costs and Benefits of Additional Removal

The NRC staff disagrees with DOE Idaho's statement that the peak dose can only be reduced up to 60 percent because only 60 percent of the activity is present in the tanks. The sand pad inventory has a negligible impact on the peak groundwater all-pathways dose, because Tc-99 is the only significant dose contributor at the time of peak dose, and the inventory of Tc-99 (and I-129, which peaks earlier) is negligible in the sand pad. The metric DOE Idaho used to quantify the costs of additional radionuclide removal in the tanks (e.g., cost per reduction in dose to the individual expected to receive the peak annual dose over a 50-year time period) may not be appropriate.

Regarding DOE Idaho's statement that typical average doses to the public from natural sources and medical treatment in the range of 3–4 mSv/yr [300–400 mrem/yr] make it impractical to attempt additional waste removal that would only reduce the estimated dose from the TFF by a small amount, NRC staff agrees that the magnitude of the maximum dose reduction {5×10^{-3} mSv/yr [0.5 mrem/yr] for a member of the public in the conservative or compliance case groundwater all-pathways scenario} should be considered in evaluating the practicality of additional removal. Although NRC's current regulations are based on a linear-no threshold

Table 6. Comparison of Various Tank Cleaning Alternatives

Performance Objective	Dose Limit*	Current System†	New Technology‡		Complete Tank Removal§	
BENEFITS (expected dose or percent reduction in dose)	mrem/yr	expected dose mrem/yr	expected dose mrem/yr	percent dose reduction	expected dose mrem/yr	percent dose reduction
All-pathways dose to public	25 mrem/yr	0.46 mrem/yr	0.00 mrem/yr	−100%	0 mrem/yr	−100%
Acute intruder for drilling	500 mrem/yr	152 mrem/yr	104 mrem/yr	−31%	0 mrem/yr	−100%
Acute intruder for construction	500 mrem/yr	0.23‖ mrem/yr	0.23 mrem/yr	−0%	0 mrem/yr	−100%
Chronic intruder for postdrilling	100 mrem/yr	25 mrem/yr	19 mrem/yr	−25%	0 mrem/yr	−100%
Chronic intruder for construction	100 mrem/yr	3.15 mrem/yr	3.15 mrem/yr	−0%	0 mrem/yr	−100%
COSTS (dose or dollars) Protection of individuals during operations (rem/yr)	5 rem/yr	expected value 0.03 rem/yr	additional cost 0.03 rem/yr^		additional cost 1.1 rem/y	
Total worker dose (person-rem)		7.15 person-rem	7.15 person-rem^		9,000 person-rem	
Cost for Cleaning 11 Large Tanks (dollars)		40 million#	40 million		5 billion	

*11 mrem/yr = 0.01 mSv/yr

†Preferred, current system is washball, directional nozzles, mechanical arms, and steam jets. Total cost of development and deployment estimated as $40 million.

‡The new technology is unknown. The maximum performance or possible effectiveness of a new technology is listed to maximize the potential benefit (i.e., the new technology is assumed to eliminate the risk associated with the tanks through additional removal or stabilization). No cleaning is assumed for sand pad or piping.

§Complete tank removal would result in very small exposures to the public. These very small exposures are shown as zero values in the table.

‖DOE Idaho (2005) Table 22 has an error for the chronic intruder construction scenario. The value of 0.23 should be 3.15 as indicated in Table 18.

^Worker dose is assumed to be the same as the current technology. Exposures could occur from reconfiguration of TFF equipment and disturbance of contaminated TFF soils; additional cleaning of the large tanks, and re-sampling residual waste in the tanks. Worker doses for this alternative are expected to be higher than the current technology. Costs associated with worker dose associated with development and implementation of a new technology are minimized for the purposes of this analysis.

#This cost includes the additional cost of approximately $1 million/tank for the remaining four tanks to be cleaned or an additional $4 million.

(LNT) model, the scientific community has raised concerns regarding the use of collective dose and the LNT model in radiological optimization analyses. Use of thresholds in calculating collective dose; a cap on lifetime or yearly individual doses or total collective dose that can reasonably be used in cost-benefit analyses (HPS, 2004; EC, 1999); and use of comparisons against background radiation and risk have been proposed as modified metrics or caveats in the use of collective dose in radiological optimization analyses. However, the use of an individual risk estimate to the average member of the critical group or maximally exposed individual appears to be an accepted approach for use in cost-benefit analyses.

Considering the fact that the groundwater all-pathways dose is expected to be less than 1 mrem/yr to a member of the public at the maximum point of exposure (less than 5.0×10^{-7} excess cancer risk per year of exposure; ICRP, 1990) and that the risk of much higher doses is low (see discussion in Chapter 4), additional radionuclide removal appears to have minimal benefit. On the other hand, costs associated with development and deployment of a new technology are expected to double the financial cost and worker dose compared to completing tank cleaning activities with the current technology. Complete tank removal is cost prohibitive.

3.8.4 Comparison of Costs for Other TFF Regulated Activities

To provide context for the assessment of costs and benefits associated with attempting additional removal of TFF waste, NRC staff considered cost and benefits of remedial activities associated with historical TFF contamination [e.g., CERCLA activities associated with the TFF, including RI/BRA risk estimates, Corrective Measures Study/Feasibility Study remedial alternative analysis, and the proposed plan (DOE Idaho, 2006e–g].

For comparison, total projected costs of deployment of a final CERCLA remedial action to address existing contamination of soils and groundwater at the TFF resulting from previous releases at the TFF (see Section 1.1.2.5) are expressed as net present value in fiscal year 2006 dollars, for alternatives that meet the threshold criteria, and range from $9 million for Alternative 2b to $44.5 million for a combination of Alternatives 3a and 5 described below (DOE Idaho, 2006f). Alternative 2b is the preferred alternative (DOE Idaho, 2006g).

Alternatives include the following:

- Alternative 1—Institutional Controls, Operations and Maintenance, and Monitoring. This will be referred to as limited action and meets the intent of the no-action alternative.

- Alternative 2a—Institutional Controls, Monitoring, Excavation, and Containment by 2012.

- Alternative 2b—Institutional Controls, Monitoring, Excavation, and Containment by 2035. Capping would be implemented in phases with a low-permeability asphalt cover to control infiltration by 2012 and final capping by 2035 when infrastructure constraints are removed.

- Alternative 3a—Institutional Controls, Monitoring, Excavation, Source Removal, and Containment by 2012.

- Alternative 3b—Institutional Controls, Monitoring, Excavation, Source Removal, and Containment by 2035. The remedy would be implemented in phases with a low-permeability asphalt cover to control infiltration by 2012 and source removal, and final capping by 2035 when infrastructure constraints are removed.

- Alternative 4a—Institutional Controls, Monitoring, Excavation, Source Treatment, and Containment by 2012.

- Alternative 4b—Institutional Controls, Monitoring, Excavation, Source Treatment and Containment by 2035. The remedy would be implemented in phases with a low-permeability asphalt cover to control infiltration by 2012 and source treatment and final capping by 2035 when infrastructure constraints are removed.

- Alternative 5—Contingent SRPA Pump and Treat for Cleanup. This will be referred to as contingent pump and treat. It would only be implemented in approximately 2077 if Alternatives 2a, 2b, 3a, 3b, 4a, or 4b had already been implemented and determined through groundwater monitoring not to be sufficiently protective of the aquifer.

The risks associated with the CERCLA remedial alternatives listed above are as high as 2.0×10^{-2} excess cancers from surficial soil contamination, primarily from Cs-137 to individuals working in the vicinity of the TFF today (DOE Idaho, 2006g). Groundwater is also currently contaminated above applicable regulatory requirements established by the EPA (maximum contaminant levels (MCLs)), and the risk to the public is expected to continue beyond the institutional control period (beyond 2095 evaluated in the CERCLA risk assessment). The costs of additional remediation of soil and groundwater are in most cases lower than the costs associated with TFF cleaning using the current technology, while the benefit of additional soil and groundwater remediation is expected to be higher and more certain. It is significant to note that these costs and benefits are associated with final actions related to soils and groundwater at the TFF. These costs do not include the costs of interim actions currently in place, nor do they consider additional source areas at INTEC that are addressed separately.

3.9 NRC Review and Conclusions—Criterion Two

The NRC staff had a number of RAIs with respect to the sand pad inventory, tank sampling, and criteria to demonstrate removal to the maximum extent practical. DOE Idaho addressed many of these concerns. NRC staff concludes that the inventory developed for the PA appears to be bounding for the tanks. DOE Idaho provided additional information to provide confidence in the solids analytical results for the cleaned tanks. Although variability for certain radionuclides (e.g., Sr-90 and Cs-137) is expected from tank to tank, DOE Idaho used a reasonable approach to estimate the inventory for uncleaned tanks. DOE Idaho also used a reasonable approach to evaluate the uncertainty associated with the sand pad inventory. The uncertainty in the identification of potential HRRs in the sand pad is addressed by NRC staff's consideration of TFF monitoring well data that provide trend information on SRPA impacts from historical releases. NRC staff finds reasonable assurance that Criterion Two of the NDAA can be met.

Mechanical and chemical cleaning technologies were evaluated. Emphasis was placed on the specific chemical and physical form of the TFF wastes when evaluating the available technologies. Because the TFF waste is acidic, bulk chemical cleaning is not deemed to be practical. Cleaning operations have demonstrated that the mechanical processes selected for bulk waste removal are effective. Complete tank removal would essentially eliminate potential annual doses to the public, including inadvertent intruders. However, tank removal would be economically impractical and would result in relatively large worker exposures compared to the current technology.

Approximately 90 percent of the solid residual heels are estimated to have been removed using the washball and directional nozzle technology. Removal efficiencies considering the cumulative historical inventory stored at the TFF of 3.2 million TBq (87 million Ci) and a final inventory of approximately 955 TBq [25,800 Ci] estimated at closure are over 99.9 percent.

The following assumption was made in assessing conformance with Criterion Two:

- Inventory estimates for the large tanks that have not been cleaned (WM–187 through WM–190) are not significantly underpredicted (i.e., similar or better waste retrieval will be achieved than is currently assumed by DOE Idaho).

The NRC staff conclusions with respect to Criterion Two are the following:

There is reasonable assurance that Criterion Two of the NDAA can be met because:

- The estimated inventory developed for the tanks and sand pad in the performance assessment and validated through sampling for the tanks in the waste determination is reasonable for the purpose of evaluating compliance with NDAA criteria.

- NRC staff has confidence that HRRs have been or will be removed to the maximum extent practical based on an evaluation of DOE's selection of HRRs; DOE's selection, implementation and effectiveness demonstration for its preferred cleaning technology; and NRC staff's evaluation of the costs and benefits of additional removal.

4 CRITERION THREE (A) AND THREE (B)

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

(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and

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

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

(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations;

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

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

DOE must determine whether the waste that is the subject of the waste determination exceeds concentration limits for Class C LLW that are provided in 10 CFR 61.55 to determine whether Criterion 3(A) or Criterion 3(B) provided above is applicable. In response to NRC RAI 17 (DOE Idaho, 2006b), DOE Idaho determined that the final wasteform disposed of in the TFF is less than Class C based on its application of draft NRC guidance on concentration averaging (70 FR 74846). NRC's evaluation of DOE Idaho's assessment of TFF waste classification is presented in Section 4.1. Whether the waste is greater than or less than Class C, DOE Idaho must demonstrate that the waste will be disposed of in compliance with the performance objectives set out in Subpart C of 10 CFR Part 61 (NDAA Criterion (3)(A)(i) and (3)(B)(i) listed above). Additionally, Criterion 3(A)(ii) or 3(B)(ii) provides the State of Idaho with a role in approving closure plans or permitting the disposal facility. The State of Idaho oversees the closure of the TFF at INTEC to ensure that waste is disposed of in accordance with Hazardous Waste Management Act (HWMA)/Resource Conservation and Recovery Act (RCRA) requirements. DOE Idaho is currently seeking approval for clean closure of the tanks at INTEC (DOE Idaho, 2003b), which would allow DOE Idaho to close the TFF without a state permit. If the waste is greater than Class C, an additional requirement (i.e., Criterion 3(B)(iii)) listed above also applies.

The performance objectives of 10 CFR Part 61, Subpart C, require assessment of protection of the general population from releases of radioactivity, protection of individuals from inadvertent intrusion into the waste, protection of individuals during operations, and evaluation of the stability of the disposal site after closure. Protection of the general population (including inadvertent intruders) is typically evaluated through a PA calculation that takes into account the relevant physical processes and the temporal evolution of the system. The NRC staff's assessment of DOE Idaho's PA (2003a) is presented in Section 4.2. The NRC staff conclusion on Criterion 3 is presented in Section 4.3. Key monitoring areas that are important to assessing compliance with 10 CFR 61, Subpart C, performance objectives are identified in Section 4.4.

4.1 Assessment of Waste Classification

As assessment of the classification of waste in accordance with 10 CFR 61.55 is required to determine if NDAA Criterion (3)(A) or (3)(B) listed in Section 4.0 applies.

4.1.1 Waste Classification

LLW intended for near surface disposal is normally classified as Class A, B, or C based on concentration limits for radionuclides listed in 10 CFR 61.55. Table 7 presents a range of waste classification results based on a comparison of activity concentrations for Tanks WM–181 (tank with the lowest inventory) and WM–182 (tank with the highest inventory) against the Class C concentration limits found in Tables 1 and 2 in 10 CFR 61.55. An attempt was made to obtain analytical samples for all HRRs (see Section 3.3) which includes all radionuclides specifically listed in Tables 1 and 2 in 10 CFR 61.55, thereby increasing confidence in the waste classification calculations. 10 CFR 61.55, Table 1, also includes a class of radionuclides (e.g., alpha-emitting transuranic radionuclides with half-lives greater than 5 years) that were also included in the list of HRRs. However, another class of radionuclides listed in 10 CFR 61.55, Table 2 (i.e., short-lived radionuclides that have half-lives less than 5 years) is eliminated from HRR consideration. The elimination of this class of radionuclides from the HRR list has no material impact on waste classification, because the waste could not be classified as greater than Class C based on the concentrations of these radionuclides. A footnote to 10 CFR 61.55, Table 2, indicates that there are no limits established for nuclides with half-lives less than 5 years in Class B or C wastes and that the waste shall be Class B (or less) unless the concentrations of other nuclides dictate that the waste is Class C or greater independent of these nuclides. The TFF waste addressed in this TER contains a mixture of long- and short-lived radionuclides; therefore, 10 CFR 61.55 (a)(5) was applied to determine waste classification. Also, 10 CFR 61.55(a)(7) for mixtures of radionuclides and 10 CFR 61.55(a)(8) related to concentration averaging are relevant to the waste classification calculations. These citations are reproduced here for additional background information on the waste classification calculations.

> 10 CFR 61.55(a)(5), "*Classification determined by both long- and short-lived radionuclides*. If radioactive waste contains a mixture of radionuclides, some of which are listed in Table 1, and some of which are listed in Table 2, classification shall be determined as follows: (i) If the concentration of a nuclide listed in Table 1 does not exceed 0.1 times the value listed in Table 1, the class shall be determined by the concentration of the radionuclides listed in Table 2. (ii) If the concentration of a nuclide listed in Table 1 exceeds 0.1 times the value listed in Table 1 but does not exceed the value in Table 1, the waste shall be Class C, provided the concentration of nuclides listed in Table 2 does not exceed the value shown in Column 3 of Table 2."

> 10 CFR 61.55(a)(7), "*The sum of the fractions rule for mixtures of radionuclides*. For determining classification for waste that contains a mixture of radionuclides, it is necessary to determine the sum of fractions by dividing each nuclide's concentration by the appropriate limit and adding the resulting values. The appropriate limits must all be taken from the same column of the same table. The sum of the fractions for the column must be less than 1.0 if the waste class is to be determined by that column. Example: A waste contains Sr-90 in a concentration of 50 Ci/m3 and Cs-137 in a concentration of 22 Ci/m3. Because

the concentrations both exceed the values in Column 1, Table 2, they must be compared to Column 2 values. For Sr-90 fraction 50/150=0.33; for Cs-137 fraction, 22/44=0.5; the sum of the fractions=0.83. Because the sum is less than 1.0, the waste is Class B."

10 CFR 61.55(a)(8), "*Determination of concentrations in wastes*. The concentration of a radionuclide may be determined by indirect methods such as use of scaling factors which relate the inferred concentration of one radionuclide to another that is measured, or radionuclide material accountability, if there is reasonable assurance that the indirect methods can be correlated with actual measurements. The concentration of a radionuclide may be averaged over the volume of the waste, or weight of the waste if the units are expressed as nanocuries per gram."

As stated above, because a mixture of radionuclides comprises the residual waste at the TFF, a sum of fractions approach is used to determine the waste classification as provided in 61.55(a)(7). TFF waste contains relatively large (with respect to waste classification) quantities of transuranics, while 10 CFR 61.55, Table 2, radionuclides (e.g., fission products) are insignificant to the sum of fractions and are not specifically listed in Table 7 below. Furthermore, because most of the activity is in the solid residual heel, the liquid activity contribution to the sum of fractions is considered insignificant (i.e., the liquid residual heel is much less radioactive than the solid residual heel) and is not specifically listed in Table 7 below.

The waste concentrations used for comparison against Class C limits consider the volume or weight of the wasteform as provided in 61.55(a)(8). NRC's draft guidance on concentration averaging (70 FR 74846) provides an acceptable approach for averaging waste over stabilizing materials (e.g., grout used to solidify or "encapsulate" the waste). The stabilized wasteform in the INL tanks is a formulation of cement, fly ash, fine aggregate, ground blast furnace slag, and water (see Appendix C in DOE Idaho, 2005; and revised formula in clarifying RAI 3 in DOE Idaho, 2006a) mixed with waste to produce a solidified wasteform to achieve stability of the waste for disposal (see Section 4.2.5 and 4.2.17). An engineered grout pour and "encapsulation"[11] pour will be used:

- To move remaining residual heels toward jet pumps to remove as much residual material from the tank as is reasonably achievable (engineered grout pour)

- To cover the waste displaced by the engineered grout pour with enough additional grout to provide stability of the wasteform ("encapsulation" pour)

Complete mixing of the waste with a large volume of grout is desirable as it may provide additional stability of the wasteform and may help ensure that future releases from effluents are maintained ALARA (e.g., it may help to reduce the magnitude and maximize the timing of

[11]DOE Idaho used the term "encapsulation" to describe the additional grout placement on top of the engineered grout pour. The term encapsulation means something different in various applications. For example, encapsulation in the Branch Technical Position (BTP) on Concentration Averaging and Encapsulation was used in the context of surrounding discrete sources (e.g., sealed source), with stabilizing material for the purposes of radiation shielding and to provide a recognizable, nondispersible wasteform (NRC, 1995). The term encapsulation was used in the draft guidance for concentration averaging for waste determinations (70 FR 74846) reproduced in the draft SRP (NRC, 2006b) to mean the averaging of waste over stabilizing materials needed to solidify waste to produce a recognizable wasteform and to prevent dispersion of radioactivity into the environment.

releases to the environment due to the chemical and physical barrier afforded by the stabilization materials). In the case that the waste is completely mixed with the stabilizing material, the entire wasteform (excluding the wasteform container) may be used for the purposes of waste classification. However, for TFF waste, minimal mixing with the grout used in the engineered grout pour is expected to occur.

The draft guidance (NRC, 2006b) states that for waste that is not homogeneously mixed with the stabilizing material, credit can be taken for the amount of grout needed to stabilize the waste and that generally the unstabilized-to-stabilized waste concentration should be within a factor of ten. DOE Idaho's response (DOE Idaho, 2006a) to NRC's RAI 17 (NRC, 2006a) focused on the following points to classify the waste as meeting Class C concentration limits: (i) intruder analyses show that 10 CFR 61.42 requirements can be met (waste classification is primarily based on risk to inadvertent intruders), (ii) a grout volume of more than 10 times the volume of waste is needed to facilitate additional waste removal, and (iii) a grout volume of more than 10 times the volume of waste is needed to stabilize the waste.

TFF waste is not recalcitrant and is amenable to removal through displacement and lifting by the engineered grout pours toward jet pumps and additional grout may be needed to stabilize any waste that is able to be lifted from the tank floor. Therefore, using this approach to remove additional waste from the tank, the waste residuals may, in fact, be distributed in a larger volume of grout than 10 times the mass or volume of the waste. However, DOE Idaho took credit for 85 m^3 [3×10^3 ft^3] of grout expected to be used in an engineered pour for the purpose of pushing and funneling remaining residual heels toward the jet pump and for another 33 m^3 [1.2×10^3 ft^3] of grout used to stabilize or "encapsulate" the waste. DOE's approach was based on the volume of grout needed to conduct the engineered grout pour, not the volume of grout needed to simply stabilize the waste. Because the waste is not expected to be well mixed in the engineered grout and "encapsulation" pours, it is not appropriate to use the total volume of grout needed for these pours to determine the waste classification (applying Category 1 of the draft guidance). Sufficient justification for why 118 m^3 [4.2×10^3 ft^3] of grout is needed to simply stabilize the waste under Category 2 of the draft guidance was also not provided. The estimated 85 m^3 [3×10^3 ft^3] of grout for the engineered pour is just slightly higher than the volume of grout needed to meet Class C concentrations (see Figure 15). The NRC staff is concerned that DOE Idaho used an approach that could support the use of an excessive amount of grout for the purpose of waste classification without sufficient justification.

In the case of the large tanks at INL, DOE Idaho did not exhaust all acceptable methods provided in the draft guidance for averaging the waste inventory over a larger volume of grout (10 times the mass or volume of the waste) in its calculation of the sum of fractions which resulted in a value of 14 for all tanks. Most significantly, DOE Idaho did not take credit for the volume or mass of liquid residuals which contributes most of the residual waste volume (about 80 percent) in calculating an acceptable volume of stabilizing grout. Consideration of the liquid residual waste volume increases the allowable amount of stabilizing grout by up to a factor of five if ten times the volume of residual waste is used.[12] DOE Idaho could also have taken credit

[12]The draft guidance allows the inventory to be averaged over an amount of stabilizing grout equivalent to ten times the volume or mass of residual waste. Because the density of the grout (2.1 g/cm^3) is higher than the density of the waste (1.4 g/cm^3), use of an amount of grout equivalent to ten times the volume of waste allows a greater amount of grout to be used to stabilize the waste than if an amount of grout equivalent to ten times the mass of waste is used.

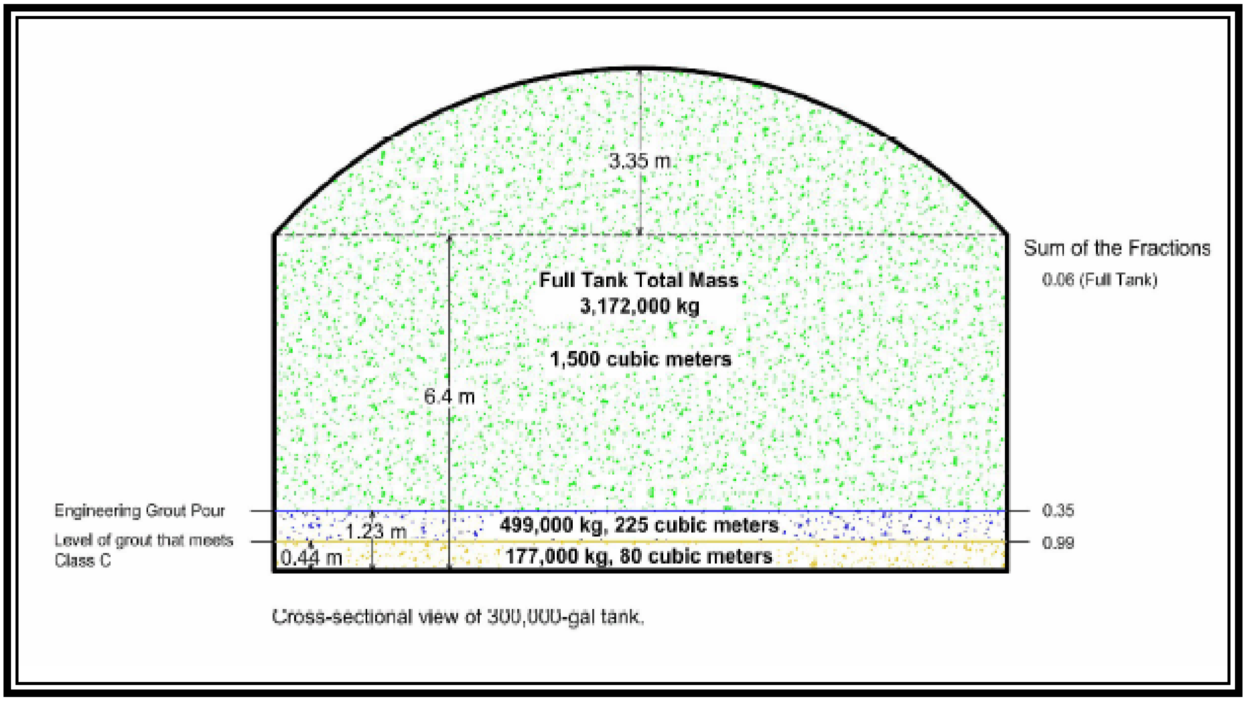

Figure 15. Representation of Thickness of Grout Needed to Meet Class C Concentration Limits (from DOE Idaho, 2005)[13]

for a portion of the large stainless steel tank walls and floor. When these credits are included, NRC staff calculations show a sum of fractions of 0.6 for Tank WM–181, the tank with the lowest inventory, and a sum of fractions of 2.8 for Tank WM–182, the tank with the highest inventory (see Table 7, column 4). If stabilizing grout in the amount of ten times the volume of waste is assumed in the waste classification calculations for all radionuclides, including transuranics that have Class C concentration units expressed in activity per unit mass (see Table 7, column 2), the range of sum of fractions is even lower, from 0.3 to 1.5 (see Table 7, column 5).

Another source of uncertainty in the waste classification is related to the inventory estimated by DOE for the large tanks. In its response to RAI 17 (DOE Idaho, 2006a), DOE Idaho stated that the large tank inventory calculations are likely overestimated because the volume of interstitial liquids within the solid residuals may be as high as 75 percent. DOE Idaho calculated the large tank inventory by estimating the solid volume through kriging methods and multiplying this volume by the WM–183 solid concentration from sampling. Based on data provided by DOE Idaho in response to RAI 17 (DOE Idaho, 2006a), the inventory may be overestimated by up to

[13]This figure provided in the draft waste determination (DOE Idaho, 2005) illustrates the sum of fractions for various levels of grout. For example, if a tank is filled to its capacity of grout, the sum of fractions with respect to Class C concentrations is only 0.06. However, the draft guidance for concentration averaging for waste determinations proposes averaging over an amount of grout needed to stabilize the waste if the waste is not homogeneously mixed with the stabilizing grout (70 FR 74846). NRC calculated the acceptable volume of grout to stabilize the waste and presented the corresponding sum of fractions for that amount of grout in Table 7.

Table 7. Comparison of Tank Waste Concentrations Against Class C Limits*

Radionuclide	Class C Limit Concentration (Ci/m^3 or nCi/g)	Unstabilized Waste Concentration (Ci/m^3 or nCi/g)	Waste Concentration with Credit Taken for Stabilizing Grout and Tank (Ci/m^3 or nCi/g)	
Radionuclides (Long-lived)				
C-14	8 Ci/m^3	5.5×10^{-6} Ci/m^3	1.0×10^{-8} Ci/m^3	
Tc-99	3 Ci/m^3	0.86 Ci/m^3	0.016 Ci/m^3	
I-129	0.08 Ci/m^3	8.7×10^{-4} Ci/m^3	1.6×10^{-5} Ci/m^3	
Ten Times the Volume Versus Ten Times the Mass of Waste is Also Provided for Transuranics (TRU) in this Table for Comparison (Ten Times the Volume Leads to Lower Concentrations Due to the Density Difference of Waste and Grout)			10 × mass of waste	10 × volume of waste
Alpha-emitting Transuranic (TRU) nuclides with half-life greater than 5 years†	100 nCi/g	1.4×10^{4} nCi/g	58–279 nCi/g	29–144 nCi/g
Pu-241†	3,500 nCi/g	1.6×10^{4} nCi/g	65–310 nCi/g	32–160 nCi/g
Cm-242†	20,000 nCi/g	1.1 nCi/g	0.004– 0.02 nCi/g	0.002–0.01nCi/g
Total Sum of Fractions		**147**	**0.6–2.9**	**0.3–1.5**
Classification		**> Class C**	**< Class C to > Class C**	**< Class C to > Class C**

*Tank WM–181 has the lowest sum of fractions, while Tank WM–182 has the highest sum of fractions. The calculations are based on a 0.5-cm tank wall thickness (a 10-in wall height is assumed) and 0.8-cm tank bottom thickness. The density of stainless steel is assumed to be 8 g/cm^3. Grout in the amount of 10 times the volume of the waste (both liquid and solid) is used for those radionuclides with concentration limits expressed in units of activity per unit volume, while grout in the amount of 10 times the mass of the waste is used for those radionuclides with concentration limits expressed in units of activity per unit mass. Alternative concentrations are provided for those radionuclides with concentration limits expressed in units of activity per unit mass to show what the concentration would be if 10 times the waste volume was used in the calculation instead of 10 times the mass. This approach allows for a greater amount of grout to be used for averaging due to the density difference of the waste, 1.4 g/cm^3 versus 2.1 g/cm^3 for the grout.

†Cm-244 is the only additional radionuclide with a significant inventory (with respect to the sum of fractions) that was not specifically identified by DOE Idaho under the category "alpha-emitting TRU with half-life greater than 5 years".

Other radionuclides identified by DOE Idaho with significant activities included in this category are Am-241, Np-237, Pu-238, Pu-239, and Pu-240. There was a discrepancy between the engineering design file (EDF) for WM-183 (Portage, 2005c) and the draft waste determination (DOE Idaho, 2005) with respect to the Pu-238 concentration. The higher value of 9.99E-03 pCi/g in the EDF data tables was used rather than the value of 9.23E-03 used in the waste determination.

a factor of three if a smaller volume of solid residual (or alternatively a lower bulk waste density) is used in the inventory calculation. Taking into consideration these factors, the sum of fractions is expected to be more than a factor of 15 lower than DOE Idaho's estimate of 14 for Tank WM–182, the tank with the highest inventory. Thus, the waste classification for all large tanks could reasonably be found to meet Class C concentrations if this assumption regarding the waste inventory is made.

In response to RAI 17 (DOE Idaho, 2006b), DOE Idaho also provided waste classification calculations for the sand pad. In the case of the sand pads, grouting the vault serves to "encapsulate" or limit the dispersal of contamination into the environment by providing a hydraulic barrier to limit contact of infiltrating water with radiological constituents in the sand pad. Additionally, the grouted tank located above the sand pad limits the infiltration of water into the sand pad. The concrete vault underneath the sand pad provides an additional chemical barrier that limits the release of contamination from the sand pad. It can be easily demonstrated that 10 times the volume, 230 m^3 [8.1×10^3 ft^3], or mass of the sand pad is more than enough grout to meet Class C concentration limits (only a few inches of grout from the vault floor beneath the sand pad or above the sand pad are needed to "encapsulate" the sand pad and meet the Class C limits). The sum of fractions for the unstabilized sand pad is relatively low at a value of 1.7. Additionally, the limited activity in the piping and smaller 100-m^3 [30,000-gal] tanks makes it also relatively easy for DOE Idaho to demonstrate that the piping and 100-m^3 [30,000-gal] tanks can meet Class C limits.

Based on this analysis, NRC staff concludes that it is reasonable for DOE to find that the residual waste in tanks, sand pads, and piping that is the subject of this waste determination does not exceed Class C concentration limits. However, NRC cannot confirm that all large tanks contain waste that is within Class C limits due to the uncertainty in the residual waste inventory. Section 3116(a)(3)(B)(iii) requires DOE to consult with the NRC in its development of disposal plans in cases where the waste exceeds Class C concentration limits. In the draft waste determination for the INL TFF, DOE stated its intent to "take full advantage of the consultation process established by Section 3116" and requested NRC to "identify what changes, if any, it would recommend to DOE's disposal plans" as described in the draft waste determination. NRC has reviewed DOE's disposal plans for TFF waste as part of the extensive consultation process that is documented in this TER, thereby satisfying the requirements of Section 3116(a)(3)(B)(iii). Consequently, no additional DOE consultation with the NRC is required for tanks containing residual waste that could exceed Class C concentrations. Idaho DEQ has also stated its view that DOE's extensive consultation with NRC meets the additional requirements for waste that could be greater than Class C (Idaho DEQ, 2006).

4.2 Performance Assessment to Demonstrate Compliance With Performance Objectives

For non-HLW determinations, DOE normally develops a PA to demonstrate that dose-based regulatory criteria found in 10 CFR Part 61, Subpart C, can be met. PA components include the evaluation of potential initiating events (both natural and anthropogenic) that can cause releases of radioactive material into the environment, estimates of the release rates of radionuclides into the environment, modeling the fate and transport of radionuclides in the environment, and evaluation of the potential pathways of exposure and consequences associated with these exposures to human health. The PAs submitted by DOE to support non-high-level waste determinations have included a collection of integrated process models to demonstrate compliance with performance objectives in 10 CFR Part 61, Subpart C.

Various approaches to PA calculations (e.g., deterministic, probabilistic) have their advantages and disadvantages. A deterministic approach can be very valuable when compliance can be easily demonstrated with parameters and models that clearly tend to overpredict the potential risk posed by the disposal facility. These type of analyses require little support of model and model parameters and thus can save a lot of time and money. However, compliance demonstration with simple deterministic models can be difficult for evaluations that assess compliance over periods that span tens of thousands of years in complex, and unique engineered and natural systems that are hard to represent conceptually, with models that have many interdependent parameters with large or unknown uncertainty, and models that have results that respond in a non-linear or unpredictable fashion within a reasonable range of parameter space. A probabilistic approach can have distinct advantages when there are a number of uncertainties that may significantly influence the results of a PA.

It is important to note that model support (i.e., data or information that supports the model or parameters used in the model) is necessary to provide confidence in the predictive capability of the PA model being evaluated. Because of the long time periods involved with most PA analyses, PA models cannot be validated in a traditional sense. However, the results of laboratory and field experiments, monitoring data, natural analogs, expert elicitation, and supporting sub-models that also have adequate support can provide multiple, supporting lines of evidence for the results of the PA model. The amount of model support provided should be commensurate with the risk reduction being provided by the natural and engineered system. Thus, a combination of approaches is often necessary to ensure key processes and parameters are identified.

4.2.1 Summary of Performance Objectives and Results

NDAA Criterion 3 states the following:

(3) (A) does not exceed concentration limits for Class C low-level waste as set out in Section 61.55 of title 10, Code of Federal Regulations, and will be disposed of–
- *(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations;*
- *(ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this Section; or*

(B) exceeds concentration limits for Class C low-level waste as set out in Section 61.55 of title 10, Code of Federal Regulations, but will be disposed of–
- *(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations;*
- *(ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this Section; and*
- *(iii) pursuant to plans developed by the Secretary in consultation with the Commission*

DOE Idaho must demonstrate that waste will be disposed of in compliance with the performance objectives in 10 CFR Part 61, Subpart C. The primary dose-based metric in Subpart C regulations is an annual dose limit to members of the public found in 61.41 (see text of regulation below). While 10 CFR 61.42 (see text of regulation below) provides requirements for protection of individuals from inadvertent intrusion after active institutional controls are removed, no dose limits are specified, and the time when active institutional controls are assumed to be removed is not specified.

The specific performance objectives in 10 CFR Part 61, Subpart C, that must be met in order to meet NDAA Criterion 3 include the following:

> 10 CFR 61.41, "*Protection of the general population from releases of radioactivity*. Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable."

The 0.25-mSv/yr (25-mrem/yr) limit applies to the post-closure period of a disposal facility. The use of total effective dose equivalent (TEDE) in lieu of whole body dose (due to use of older, whole body, and newer, TEDE and ICRP dose methodologies) is discussed in Section 4.2.10.

> 10 CFR 61.42, "*Protection of individuals from inadvertent intrusion*. Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."

In the Draft Environmental Impact Statement for Part 61 (NRC, 1981), NRC used a 5-mSv [500-mrem] annual dose limit to an acute inadvertent intruder to establish the concentration limits and other aspects of the waste classification system. This limit will be used in evaluating compliance with 10 CFR 61.42. While 10 CFR 61.42 does not specify a time when active institutional controls are assumed to be removed, the regulations in 10 CFR 61.59(b) specify that institutional controls may not be relied upon for more than 100 years. Thus, this regulation provides a basis for the requirement of a 100-year active institutional control period and these controls are assumed to fail in a site-specific intruder analysis performed by DOE to demonstrate compliance with 10 CFR 61.42.

> 10 CFR 61.43, "*Protection of individuals during operations*. Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable."

This performance objective applies to both the public and to disposal facility workers.

> 10 CFR 61.44, "*Stability of the disposal site after closure*. The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required."

The stability performance objective is consistent with a premise of Part 61 that the facility must be sited, designed, used, operated, and closed with the intention of providing permanent disposal and should not require long-term maintenance and care.

Generally, a 10,000-year compliance period is used to demonstrate compliance with the 10 CFR Part 61, Subpart C, performance objectives. This time period is normally sufficient to capture the peak dose from the more mobile, long-lived radionuclides and to demonstrate the influence of the natural and engineered systems in achieving the performance objectives (NRC, 2000). However, assessments beyond 10,000 years may be necessary to ensure that radioactive waste disposal does not result in markedly high doses to future generations for certain types of waste or to ensure that overly optimistic assumptions regarding the performance of engineered or natural barriers do not mask the potential risks of long-lived constituents (e.g., Tc-99). Periods of performance shorter than 10,000 years are generally not appropriate for disposal facilities for incidental waste, because of the larger fraction of long-lived radionuclides compared to a typical commercial LLW disposal facility.

Table 8 provides an overview of the DOE Idaho PA results compared to the performance objectives in 10 CFR 61, Subpart C (10 CFR 61.41 and 61.42). Table 9 provides an overview of the DOE Idaho waste determination results in which the performance assessment results were scaled using a ratio of the current radionuclide inventory after tank cleaning versus the inventory assumed in the PA. The results are explained in more detail in the sections that follow.

4.2.2 Performance Assessment Approach and Results

In order to conduct a risk informed review of the performance assessment it is helpful to understand the overall risk context of the system and how the performance of the system is modeled. DOE Idaho used a PA to demonstrate that the performance objectives in 10 CFR Part 61, Subpart C, listed above can be met at closure for the TFF waste. The PA provides an evaluation of the types of radioactive releases that could occur as a result of disposal

Table 8. Summary of PA Modeling Results Compared to Performance Objectives

Part 61 Performance Objective	Performance Limit (mrem/yr)	DOE Idaho PA Result*† (mrem/yr)
61.41 All-pathways dose to public	25	1.9
61.42 Intruder–Acute drilling scenario	500	232
61.42 Intruder–Acute construction scenario	500	0.8
61.42 Intruder–Chronic postdrilling scenario	500	91.1
61.42 Intruder–Chronic postconstruction scenario	500	26.1

*1 mrem/yr=0.01 mSv/yr
†DOE Idaho, 2003a

Table 9. Summary of Updated Estimated Doses*† Compared to Performance Objectives		
Part 61 Performance Objective	**Performance Limit (mrem/yr)**	**DOE Idaho PA Result* (mrem/yr)**
61.41 All-pathways dose to public	25	0.5
61.42 Intruder–Acute drilling scenario	500	152
61.42 Intruder–Acute construction scenario	500	0.2
61.42 Intruder–Chronic post-drilling scenario	500	25
61.42 Intruder–Chronic post-construction scenario	500	3.2

*1 mrem/yr=0.01 mSv/yr.

†The draft waste determination (DOE Idaho, 2005) presents an updated inventory based on post-cleaning sampling results (see Section 3.1). PA (DOE Idaho, 2003a) modeling results were scaled to the new inventories for a revised estimate of the expected doses from TFF residuals.

of radioactive waste at the TFF, the transport of the contaminants released into the environment, the potential exposures to humans, and the resultant consequences from these exposures. DOE Idaho presented the results of a series of models performed sequentially (e.g., independent concrete degradation modeling, Disposal Unit Source Term-Multiple Species (DUST-MS; Sullivan, 2001) release modeling, PORFLOW groundwater modeling, and exposure assessment modeling) that were used in combination to demonstrate that 61.41 requirements could be met. Output from some of these models was incorporated directly into a downstream model used in the PA (e.g., DUST-MS model output was modified for inclusion in the PORFLOW model), and other model output was used to provide information regarding parameters to be used in subsequent modeling (e.g., concrete degradation modeling was used to justify the time to failure for the DUST-MS model). Some models were used to verify results of other models (e.g., PORFLOW results were used to verify DUST-MS release results). Data was used to calibrate other models (e.g., analytical data from percolation ponds and perched water-level data were used to calibrate the PORFLOW model). The uncertainty in model parameters was propagated through the series of computational models (e.g., the uncertainty in the inventory, transport parameters, and infiltration rate was propagated through the series of models to determine the impact of these parameters on the dose metric, peak dose to a member of the public for a groundwater all-pathways scenario). The paragraphs that follow discuss each of the major process models or calculations used in this study.

As discussed in Section 3.1, inventories for the sand pads, tanks, and piping were developed using sampling data, modeling, and other calculations. These three sources were evaluated separately to distinguish the contribution to peak dose from each source. The inventory for each source was input into the DUST-MS release model. Based on concrete degradation modeling presented in Appendix E of the PA (DOE Idaho, 2003a), the assumed times of failure were 100 years for the outer vault and 500 years for the inner tank and tank grout (see Sections 4.2.6 and 4.2.7). Releases, calculated using DUST-MS, were assumed to occur only after the assumed time of failure.

DUST-MS (Sullivan, 2001) contains different conceptual model options for source releases (e.g., rinse with partitioning, diffusion, uniform degradation, and solubility-limited release). A surface rinse model was selected for the performance assessment (DOE Idaho, 2003a). The surface rinse model accounts for partitioning between the infiltrating water and the radionuclides in the wasteform. DUST-MS was also used to model partitioning and retardation for radionuclide transport occurring in the grouted tank, sand pad, and vault floor (see conceptual model in Figure 16).

Transport of contaminants through the vadose and saturated zones was modeled with PORFLOW (ACRI, 2000). PORFLOW is a mathematical model used for the simulation of multiphase fluid flow, heat transfer, and mass transport processes in variably saturated porous and fractured media.

The public receptor was assumed to be a residential farmer who could locate a well as close as 100 m [330 ft] from the closed waste tanks (DOE, 1999). The well is assumed to be used to withdraw water for personal consumption and for watering a small garden, as well as other domestic purposes. DOE Idaho developed a FORTRAN program to convert radionuclide concentrations in environmental media into annual doses following dose methodology presented in various reports (Maheras, et al., 1997; NRC, 1977; Peterson, 1983). The all-pathways scenario assumed that a receptor received radiation doses by consuming contaminated groundwater, contaminated animal products, and contaminated leafy vegetables and produce.

The location of the potential receptor is the point of maximum exposure downgradient from the TFF [estimated to be 600 m (2,000 ft)] based on the site hydrology model used for the PA (DOE Idaho, 2003a). The contribution to the maximum groundwater concentration is assumed by DOE to be from two tanks based on the expected groundwater flow direction (southerly). A two-dimensional model oriented in the direction of groundwater flow of unit thickness containing a cross section of two tanks was employed to evaluate public exposures via the groundwater pathway (see conceptual model in Figure 17). The water pathway modeling and its use in the performance assessment will be discussed in detail in Sections 4.2.8 and 4.2.9.

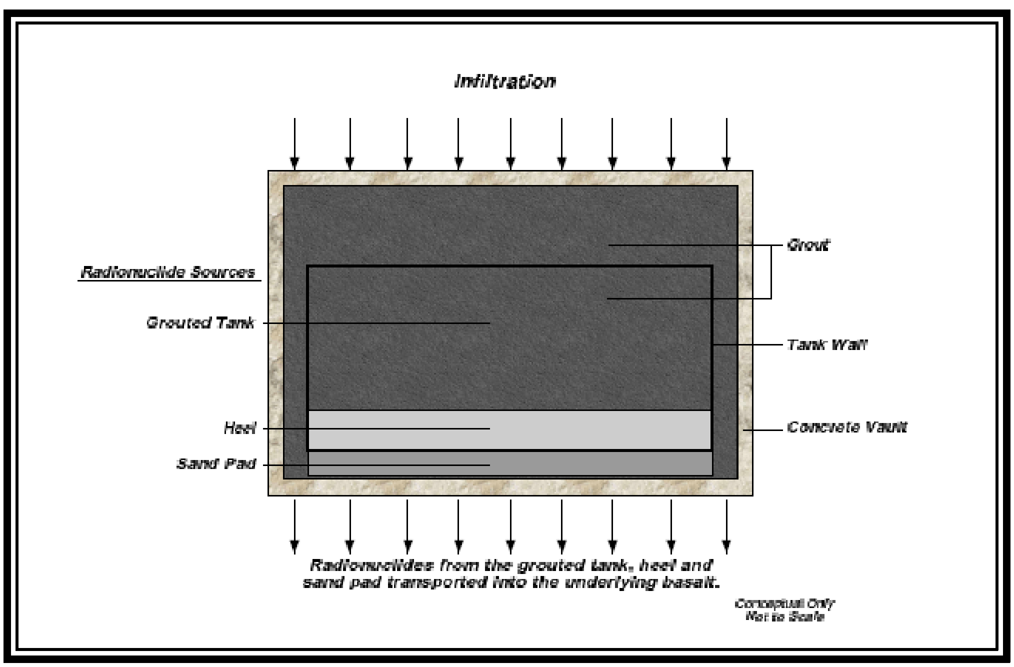

Figure 16. Conceptual Model for DUST-MS Release (From DOE Idaho, 2003a)

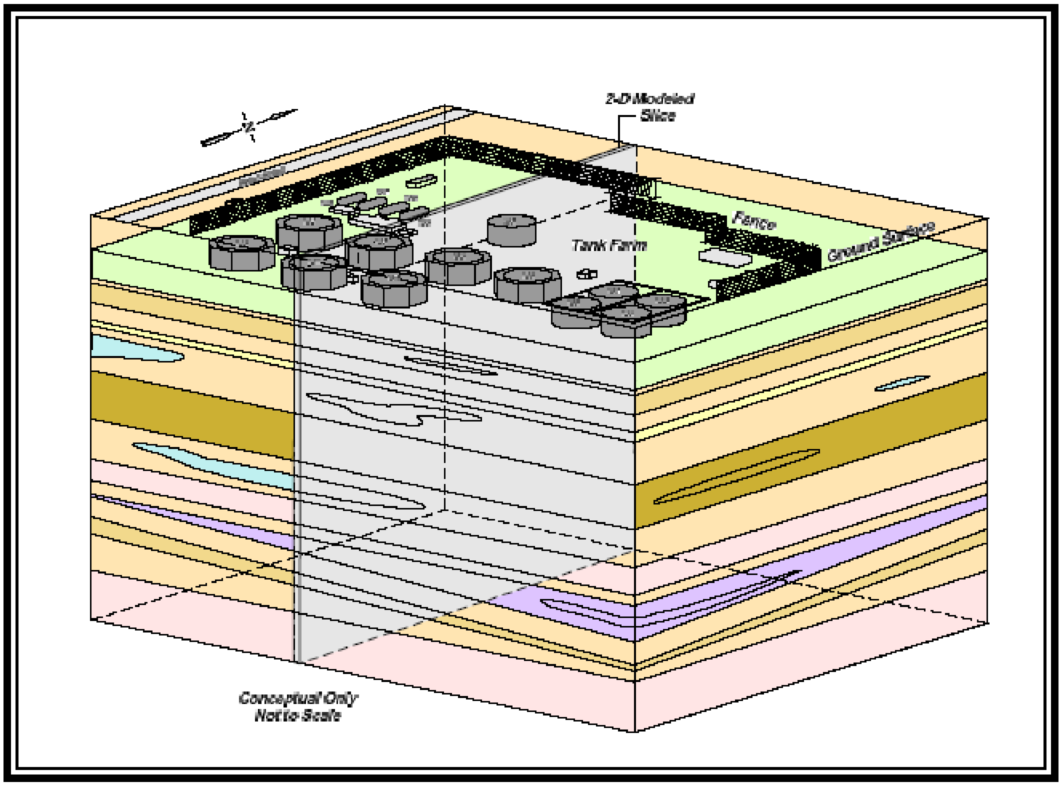

Figure 17. Conceptual Model of Groundwater Flow (From DOE Idaho, 2003a)

DOE Idaho performed analyses to investigate the sensitivity of model predictions on parameter selections (DOE Idaho, 2003a, Section 7). DOE Idaho modeled four separate scenarios: best-, realistic-, conservative-, and worst-case. For each of these scenarios, model simulations were conducted for variations in source inventories (see Section 3.1), release and transport parameters (see Sections 4.2.6, 4.2.7, 4.2.8 and 4.2.9), and infiltration rates (see Section 4.2.4 and 4.2.5). DOE Idaho used the conservative case as the compliance case to demonstrate compliance with the 10 CFR 61.41 dose-based performance objective. The sensitivity analyses using the best-, realistic-, and worst-case scenarios were intended to evaluate the potential range of doses around the conservative case (DOE Idaho, 2003a). The results of the sensitivity analysis are presented in Table 10a. The parameter values used in the sensitivity analysis are presented in Table 10b. The results of the analysis show that the most important parameter values are (i) the reducing grout K_d for Tc-99, (ii) the sedimentary interbed K_d for Sr-90, and (iii) the infiltration rate. Detailed discussion regarding the results of the sensitivity analyses can be found in the relevant sections listed above and in Section 4.2.13.

Additionally, DOE Idaho and NRC staff identified a number of specific features, events, and processes that were evaluated through additional uncertainty or sensitivity analyses, alternative conceptual models, additional qualitative analysis, analysis of monitoring data, or other means.

Some of the major features, events and processes that were evaluated are specifically discussed in this chapter and include:

- Features and properties of the disposal site that influence the degradation of engineered systems and the release of radionuclides from disposal systems (e.g., preferential flow of contaminants through cracks, fractures, or gaps between major system components such as tank walls and grout; and diffusion-dominated releases from the wasteform)

- Processes that influence the partitioning and mobility of the waste inventory (e.g., solubility versus partitioning dominated releases, effect of changes in reductive capacity of the wasteform over time, changing sorptive capacity of subsurface materials due to releases of high ionic strength wastes or low/high pH waste water into the unsaturated zone)

- Effects of surface water features on unsaturated flow (e.g, transient effect of BLR flow and flooding on contaminant flow and transport; effects of BLR on unsaturated zone flow direction and transport)

- Physical properties of surface and subsurface soils (e.g., increased hydraulic properties of disturbed alluvium and historic BLR channels, sorptive capacity of alluvium, basalt, and sedimentary interbeds)

- Physical features of the unsaturated zone that will result in preferential flow pathways (e.g., basaltic fractures and rubble zones; geologic features (volcanic vents, dikes, and fissures) or structure (sedimentary bed dip); and stress fields that affect unsaturated zone flow)

Specific features, events, and processes that were considered and eliminated by DOE Idaho (DOE Idaho, 2003a) include the following:

- Features of local flora and fauna that may impact the release of waste (e.g., deep rooting species or burrowing animals may facilitate the uptake of contaminants by humans and animals). Given the depth to waste, these features, events, and processes are expected to be bounded by other intruder analyses that have the ability to bring much larger quantities of waste to the surface(e.g., well intruder scenario).

Some of the features, events, and processes listed above impact the long-term stability of the disposal site and are also discussed in Sections 4.2.18 and 4.2.19 (e.g., flooding scenario). Inadvertent intrusion exposure scenarios are discussed in greater detail in Sections 4.2.14 and 4.2.15.

Table 10a. Sensitivity Analysis Results

Parameterization				All-Pathways Dose (mrem/yr) (yr Postclosure)			
Grout Kd	Transport Kd	Infiltration cm/yr	Inventory	129I	99Tc	90Sr/90Y	Total (yr postclosure)
Worst-Case	Worst-Case	12.4	Worst-Case	40.4 (538)	7.52 (2370)	85.8 (294)	85.8 (294)
			Conservative	15.2	6.98	85.8	85.8
			Realistic	11.7	0.630	85.8	85.8
			Best	7.76	0.38	85.8	85.8
Worst-Case	Worst-Case	4.1	Worst-Case	15.9 (607)	2.65 (5060)	15.0 (342)	15.9 (607)
			Conservative	5.97	2.46	15.0	15.0 (342)
			Realistic	4.61	0.22	15.0	15.0
			Best	3.05	0.13	15.0	15.0
Conservative	Conservative	12.4	Worst-Case	9.98 (635)	3.29 (4270)	0.12 (453)	9.98 (635)
			Conservative	3.75	3.05	0.12	3.75
			Realistic	2.89	0.28	0.12	2.89
			Best	1.92	0.16	0.12	1.92
Conservative	Conservative	4.1	Worst-Case	3.59 (890)	0.94 (1.46×10^{4})	0.006 (551)	3.59 (890)
			Conservative	1.35	0.87	0.006	1.35
			Realistic	1.04	0.08	0.006	1.04
			Best	0.69	0.05	0.006	0.69
Realistic/Best	Realistic/Best	12.4	Worst-Case	2.61 (1060)	1.62 (8100)	2.36×10^{-4} (856)	2.61 (1060)
			Conservative	0.98	1.50	2.36×10^{-4}	1.5
			Realistic	0.76	0.14	2.36×10^{-4}	0.76
			Best	0.50	0.08	2.36×10^{-4}	0.50
Realistic/Best	Realistic/Best	4.1	Worst-Case	0.87 (1960)	0.5 (2.33×10^{4})	1.75×10^{-6}(988)	0.87(1960)
			Conservative	0.33	0.46	1.75×10^{-6}	0.46
			Realistic	0.25	0.04	1.75×10^{-6}	0.25
			Best	0.17	0.02	1.75×10^{-6}	0.17

Table 10b. Sensitivity Analysis Parameters

Measured Item		Best Scenario	Realistic Scenario	Conservative Scenario	Worst-Case Scenario
Solid Radionuclide Inventory		50% reduction from worst case	25% reduction from worst case	10% reduction from worst case	Depicts sodium-bearing waste (undiluted tank-heel residual)
Liquid Radionuclide Inventory		95% reduction from worst case	80% reduction from worst case	50% reduction from worst case	Depicts sodium-bearing waste (undiluted tank-heel residual)
Infiltration		1.1 cm/yr	1.1 cm/yr	4.1 cm/yr	12.4 cm/yr
Tank Grout Sorption Coeff (m3/kg)	I	0.03	0.03	0.008	0.002
	Sr	0.006	0.006	0.003	0.001
	Tc	5	5	2.5	1
Sand Pad Sorption Coefficients (m3/kg)	I	0.001	0.001	0.001	0.001
	Sr	0.015	0.015	0.015	0.015
	Tc	0.0001	0.0001	0.0001	0.0001
Vault Sorption Coefficients (m3/kg)	I	0.03	0.03	0.008	0.002
	Sr	0.006	0.006	0.003	0.001
	Tc	0.001	0.001	0.001	0.001
UZ Long Disp (m)	Sediment	0.52	0.52	0.29	0.052
	Basalt	3.36	3.36	1.85	0.34
UZ Transverse Dispersivity (m)	Sediment	0.26	0.26	0.14	0.026
	Basalt	1.7	1.7	0.94	0.17
Sediment Sorption Coeff (mL/g)	I	5	5	0.1	0.01
	Sr	24	24	18	12
	Tc	0.1	0.1	0.01	0
Basalt Sorption Coefficients (mL/g)	I	1	1	0.1	0
	Sr	13	13	6	1
	Tc	0.24	0.24	0.01	0

4.2.3 NRC Evaluation of Performance Assessment Approach

DOE Idaho used a reasonable methodology for identifying and evaluating features, events, and processes that would affect the ability of the disposal system to meet performance objectives. DOE Idaho evaluated a large range of alternative conceptual models, performed sensitivity analyses (see Section 4.2.2), provided or analyzed additional data, or used other means to demonstrate performance objectives of 10 CFR Part 61, Subpart C could be met. The PA presents logical development of the types of exposure pathways important for receptors located at INTEC; the receptor characteristics and exposure scenarios are reasonable; and the dose limits, dose methodology, point of compliance, exposure period, and institutional control period are all acceptable. A summary of the point of compliance, compliance period, pathways, and performance objectives DOE Idaho addressed in its PA (DOE Idaho, 2003a) is provided in Table 11. Note that many of the performance objectives listed in this table are required by DOE Order and are not listed in 10 CFR Part 61, Subpart C.

To provide context for the risk-significance of information presented in the remaining sections in Chapter 4, Table 12 summarizes DOE Idaho's demonstration of compliance with the dose-based performance objective in 10 CFR 61.41. The values in the table represent factors above or below a particular value (e.g., concentration leading to performance objective or dose). The peak dose for each radionuclide is expected to occur at a different time; therefore, the peak doses for individual radionuclides are independent of one another. The first row shows the factor reduction in the maximum concentration of each radionuclide (assumes the entire inventory is present in the waste pore volume) needed to meet the performance objective of 0.25 mSv/yr [25 mrem/yr]. Rows 2 and 3 show the factor reduction in the maximum concentration attributable to various barriers. DOE Idaho's compliance case relies heavily on the Tc-99 reducing K_d (results in factor of 3 to 4 order of magnitude reduction), dilution in the vadose zone for all radionuclides (results in factor of 3 to 4 order of magnitude reduction), and transport parameters (K_ds) for Sr-90 in the sand pad and sedimentary interbeds.

4.2.4 Climate and Infiltration

Rain and snowmelt periodically infiltrate the gravelly alluvium in and around the INTEC facility. Even though average annual precipitation (22.1 cm/yr or 8.7 in/yr) is much less than the pan evaporation rate (109 cm/yr or 43 in/yr), water from snow melt or heavy rains can infiltrate rapidly prior to evaporation. Coarse surficial sediments and lack of vegetation permit a significant fraction of precipitation to enter the subsurface as infiltration. Infiltration may actually be greater due to impervious areas at INTEC which focus much of the surface runoff into gravelly areas or unlined drainage ditches (DOE Idaho, 2005e).

Several infiltration tests support high estimates of the fraction of precipitation that infiltrates the subsurface at INTEC (DOE Idaho, 2006e). Additionally, neutron moisture logging data was used to refine estimates of precipitation infiltration rates near TFF using the UNSAT-H computer model to assess soil moisture profiles and the downward wetting front associated with the 1994 spring snow melt (DOE Idaho, 2006e). A conclusion from this analysis is that the infiltration rate at and near the tank farm is larger than previously thought. The new value of 18 cm/yr [7.1 in/yr], used in recent modeling analyses related to TFF (DOE Idaho, 2006e), constitutes 85 percent of the average annual precipitation rate of 22 cm/yr [8.7 in/yr]. This infiltration rate is more than four times the infiltration rate used in the conservative or compliance case PA model (DOE Idaho, 2003a), in which rates were based on a range of values available in the literature at the time. Infiltration was expected to range from 0.41 to 12 cm/yr [0.16 to 4.9 in/yr] (DOE Idaho, 2003a;

Table 11. Summary of Performance Objectives (Comparable NRC Performance Objectives Are Highlighted in Grey)*			
Summary of Adopted Performance Objectives for the Period of Active Institutional Control			
Compliance Interval	**Pathway**	**Compliance Point**	**Performance Objective**
Period of active institutional control 2012 to 2112	All pathways (excluding radon)	Max point of impact at INL boundary	25 mrem/yr
	Air emissions (excluding radon)	Max point of impact at INL boundary	10 mrem/yr
	Radon emissions	TFF surface	20 pCi/m^2/s
	Groundwater	Max point of impact at INL boundary	4 mrem/yr
Summary of Adopted Performance Objectives for the Post-institutional Control Period			
Post-institutional control 2112 to 3012	All pathways (excluding radon)	Max dose beyond 100 m	25 mrem/yr
	Air emissions (excluding radon)	Max dose beyond 100 m	10 mrem/yr
	Radon emissions	TFF surface	20 pCi/m$_2$/s
	Groundwater	Max dose beyond 100 m	4 mrem/yr
Summary of Adopted Performance Objectives for Inadvertent Intruders			
Post-institutional control 2112 to 3012†	All pathways (excluding radon in air and groundwater)	Point of maximum dose	500 mrem (acute) 100 mrem/yr (chronic)‡

*1 mrem/yr = 1 × 10^{-2} mSv; 1 pCi = 3.7 × 10^{-2} Bq
†DOE Order requires evaluation of a 1,000-year compliance period. However, groundwater analyses were performed by DOE for 1,000,000 years to evaluate longer-lived radionuclides. This time period encompasses NRC's suggested compliance period of 10,000 years (NRC, 2006b).
‡NRC uses a performance objective of 500 mrem/yr for both acute and chronic intruder scenarios. DOE Order 435.1 uses the lower dose limit of 100 mrem/yr for the chronic intruder.

Table 12. Summary of DOE Idaho Credit for Engineered and Natural Barrier Performance*

	Tc-99	Sr-90	I-129
Minimum Total Barrier Performance Needed for Compliance†	4 orders of magnitude	9 orders of magnitude	3 orders of magnitude
Engineered Barrier (most effective of grouted tank, vault, or sand pad)	1 to 4 orders of magnitude	4 orders of magnitude‡	1 to 2 orders of magnitude
Natural System	3 to 4 orders of magnitude	8 to 9 orders of magnitude§	3 to 4 orders of magnitude
Total Barrier Performance in DOE PA Conservative or Compliance Case‖	6 orders of magnitude	12 orders of magnitude	4 orders of magnitude

*This table summarizes the credit DOE Idaho took for engineered and natural system performance in attenuating releases of Tc-99, Sr-90, and I-129, from the INTEC TFF, based on its performance assessment (DOE Idaho, 2003a). Row 1 (highlighted in grey) provides a rough factor (within an order of magnitude) reduction necessary in the waste pore water concentration to achieve levels that will meet the 10 CFR 61.41 dose-based performance objective of 25 mrem/yr. Rows 2 and 3 provide a rough factor reduction in concentration and dose attributable to various barriers as indicated.

†Row 1 is based on the maximum, possible pore water concentration. The concentration used for this calculation is recognized as being very pessimistic because actual exposure to a receptor at the maximum concentration is virtually impossible.

‡The engineered system factor for Sr-90 is based on the contribution of (i) sorption and (ii) decay in the sand pad and vault floor. The Sr-90 sorption factor (2 orders of magnitude) is calculated based on the *Kd* for Sr-90 in the sand pad. The Sr-90 decay factor (2 orders of magnitude) is based on the DOE performance assessment modeled transport times through the sand pad and vault floor (see Table 4-1, DOE Idaho, 2003a). The combined factor for the engineered barrier is, therefore, 4 orders of magnitude. It is important to note that the Sr-90 dose can be completely eliminated with more optimistic assumptions regarding barrier performance (e.g., if the tank vaults remain intact for a few hundred years, Sr-90 and other short-lived radionuclides will decay to negligible levels).

§The natural system factor for Sr-90 is based on a contribution of (i) dilution, (ii) sorption, and (iii) decay in the unsaturated zone performance assessment model (DOE Idaho, 2003a). The DOE performance assessment model results provided in Figure 21a (DOE Idaho, 2006c) shows significant dilution in the unsaturated zone (3 to 4 orders of magnitude) and virtually no dilution in the saturated zone. For simplicity, the calculated sorption factor (2 orders of magnitude) assumes that the entire plume travels through the sedimentary interbed. The Sr-90 decay factor (3 orders of magnitude) is based on the DOE performance assessment modeled transport times through the unsaturated zone which can be inferred from DOE Table 4-1 (DOE Idaho, 2003a). The combined factor for the natural system is, therefore, 8 to 9 orders of magnitude.

‖Row 4 (highlighted in grey) presents the total factor reduction in concentration DOE Idaho took credit for (includes both engineered and natural systems) in its performance assessment compliance case (DOE Idaho, 2003a).

Cecil, et al., 1992). Most of the reported values in the literature are estimates based on the amount of precipitation and best guess estimates of evapotranspiration rates for the area (DOE Idaho, 2003a).

Efforts to reduce infiltration at the TFF included installation of an impermeable polyolefin plastic cover over the surface of the TFF in 1977 to prevent water infiltration. The membrane was laid in individual sections and was drawn up and fitted around aboveground structures, and the seams were sealed. However, during the years following its installation the tank farm cover reportedly has been repeatedly damaged during construction activities. It is generally believed that the cover is no longer effective in preventing infiltration (DOE Idaho, 2006e). More recently, surface water drainage system management activities have included grading activities, construction of new ditches and providing concrete lining for old ditches, installation of a new trench drain, replacement of existing culverts with larger culverts, and construction of a large, double-lined storm-water evaporation pond east of INTEC. Asphalt has been used to cover source areas including CPP-31 (see Figure 6) at the TFF. Unpaved and gravel surfaces within and surrounding the tank farm were sealed with asphalt to prevent water infiltration and divert surface water toward the storm-water collection system (DOE Idaho, 2006e).

In 2002, two percolation ponds located immediately south of INTEC were permanently taken out of service. However, the relocation of the percolation ponds has had essentially no effect on perched water levels in the northern part of INTEC where the TFF is located. In 2004, the Sewage Treatment Plant wastewater effluent discharge was rerouted to new percolation ponds located 3.2 km [2 mi] west of INTEC, and the four wastewater infiltration trenches near the northeast corner of INTEC were permanently taken out of service (DOE Idaho, 2005e). However, an analysis of perched water levels in northern INTEC monitoring wells shows a strong positive correlation with precipitation (DOE Idaho, 2005e); therefore, the impact of these activities on contaminant flow and transport from the TFF may be minimal.

Anthropogenic sources of water include intentional clean water discharges to ground and accidental water leaks from underground water pipelines. The primary water systems include raw water, fire water, treated (softened) water, demineralized water, steam condensate, landscape watering, potable water, industrial service wastewater, and sanitary waste systems. Leaks have been discovered in the fire water and potable water pipelines (DOE Idaho, 2005e). These leaks have been repaired, but the potential exists for additional unknown leaks in the 23 km (14 mi) of underground piping at INTEC (Rodriguez, 1997). With respect to infiltration through the vault structures that hold the tanks, the vault ceilings are covered with approximately 3 m [10 ft] of soil to provide both radiation shielding and an infiltration buffer. However, in one study seepage rates into the tank vaults were found to be approximately 109,780 L/yr [29,000 gal/yr] from precipitation and lawn irrigation infiltration (DOE Idaho, 2006e). The grouted vault will provide a hydraulic barrier for infiltrating water, and recent infiltration control activities may also help shed water away from the TFF.

The DOE PA states that the TFF area will be covered with an engineered barrier under the CERCLA program (DOE Idaho, 2003a). The measured infiltration rate of 1.6 in/yr [4.1 cm/yr] for the Central Facilities Area (CFA) Landfill that has an earthen-based cover using materials from the surrounding area was considered by DOE Idaho to be an appropriate value for the conservative compliance case, because it was expected that a more robust engineered design would be implemented under the CERCLA program (DOE Idaho, 2006a). However, according to the waste determination (DOE Idaho, 2005) current plans for tank closure do not include additional engineered barriers or controls (e.g., a cover system or a cap over the TFF).

Limiting the amount of water flowing through the vaults and contacting the waste is important for limiting releases of radionuclides from the vaults, which are assumed to degrade over time. Institutional control of the present-day soil cover and any engineered covers installed in the future are assumed to be maintained for 100 years. Caps or covers, if used, will be designed to

(i) minimize water infiltration to the extent practicable, (ii) maximize flow of percolating or surface water away from the disposed waste, and (iii) resist degradation by surface geologic processes and biotic activity. The interim action for TFF soils includes the following surface water controls:

- surface water run-on diversion channels sized to accommodate a 1-in-25-yr, 24-hr storm event

- grading and surface sealing the TFF soils or sufficiently covering the TFF to divert 80 percent of the precipitation

The interim action is projected to last 8 years or until a final risk management decision is made and implemented by the agencies (DOE Idaho, 2003a). More recently, a feasibility study was completed for the TFF soils that evaluated several remedial alternatives. Remedial alternative 2b which includes an engineered cap and monitoring is the preferred alternative (DOE Idaho, 2006g).

4.2.5 NRC Evaluation—Climatology and Infiltration

DOE Idaho has not assessed the long-term performance of a robust engineered barrier that might be used to limit infiltration in the future. However, DOE Idaho did assume infiltration rates consistent with the CFA earthen cover that was expected to limit infiltration to 4.1 cm/yr [1.6 in/yr]. In the PA (DOE Idaho, 2003a) sensitivity analysis, DOE Idaho evaluated the impacts of much larger infiltration rates up to 12 cm/yr [4.7 in/yr] (DOE Idaho, 2003a), which partially addresses the impact of this assumption. However, as noted above, recent modeling suggests that infiltration rates could be higher than expected even under the worst-case scenario evaluated in the sensitivity analysis. A value of 18 cm/yr [7.1 in/yr] was used in the latest groundwater modeling analysis under the CERCLA program (DOE Idaho, 2006e). Therefore, the NRC staff concludes that the impact of increased infiltration rates or alternatively, an evaluation of infiltration controls following issuance of a final Record of Decision for TFF soils, is necessary. Recommendations related to infiltration and infiltration controls are provided in Appendix A. The impact of increased infiltration rates is also discussed further in Sections 4.2.8 and 4.2.9.

4.2.6 Engineered Barrier Degradation and Radionuclide Release

The closure strategy for the TFF at the INL involves (i) the removal of waste from the tanks, (ii) decontamination of tank vaults and piping, and (iii) confirmatory sampling and analysis. Subsequently, the tanks, vaults, and decontaminated waste piping will be grouted to minimize postclosure release of radionuclides by stabilizing the residuals in a solid matrix.

4.2.6.1 Engineered Barrier Degradation Modeling

Degradation of the various engineered barriers (i.e., grout, tanks, vaults, and piping) affects the permeability along, and the release of radionuclides through, groundwater pathways. Potential degradation mechanisms and factors that can affect permeability include initial cracks and voids; sulfate and magnesium attack; calcium hydroxide leaching; alkali-aggregate reaction; carbonation; acid attack; and corrosion of the tanks, pipes, and concrete steel reinforcement. DOE Idaho (2003a) presented a detailed analysis of engineered barrier degradation that included the effects of sulfate and magnesium attack, carbonation, and calcium hydroxide leaching on permeability. Corrosion of the reinforcement in the outer vault and localized corrosion of the tank also were modeled. The effects of acid attack and alkali-aggregate reaction on the degradation of concrete and grout were assumed to be insignificant compared to the other chemical degradation mechanisms. The results of the degradation analysis indicated that the concrete vaults turn to rubble approximately 500 years after closure, and the grout between the vault and the tank turns to rubble after 5,000 years. The analysis also indicated that the tank and the grout in the tank completely degrade and turn to rubble at approximately 40,000 years after closure. The grout associated with the piping turns to rubble after approximately 500 years if the stainless steel piping is conservatively assumed to corrode instantaneously.

4.2.6.2 Engineered Barrier Degradation Assumptions for Source Term (Release) Modeling

The results from the engineered barrier degradation modeling were abstracted into a single parameter or assumption for the DUST-MS modeling (i.e., the time to failure). DOE Idaho took a conservative approach in its PA analysis to account for engineered barrier degradation. The concrete vault and the grout between the vault wall and the tank were assumed to be completely degraded at 100 years, at which time infiltrating water was assumed to contact the radionuclides present in the sand pad and to transport these radionuclides through the sand pad and degraded vault floor to the vadose zone. At 500 years, the stainless steel tanks were assumed to have totally corroded and the grout inside the tank was assumed to have completely degraded. Also at this time, infiltrating water was assumed to contact the radionuclides in the grouted waste form and to transport radionuclides through the degraded grout, the sand pad, and the degraded vault floor to the vadose zone. Piping releases were assumed to occur in the same manner as releases from the tanks, starting at 500 years after postclosure.

The uncertainty in the time of failure of the engineered barriers is large. To provide support for the failure times assumed in the PA, DOE Idaho conducted an additional degradation analysis in which a 50 percent loss of grout was assumed to be the point of degradation failure of the engineered barriers. Table 13 compares the degradation times derived from the additional "base-case" degradation analysis, the values assumed in the PA, and the minimum and

Table 13. Summary of Degradation Analysis and PA Failure Time Assumptions (From Table 7, DOE Idaho, 2003a)

TFF System Component	PA Failure Time Assumption (yr)	Base-Case Degradation Failure Time (yr)	Minimum Degradation Failure Time (yr)	Maximum Degradation Failure Time (yr)
Vault	100	175	100	>10,000
Grout (between vault and tank)	100	3,500	500	>10,000
Piping	500	8,000	1,750	>10,000
Tank and grout inside the tank	500	8,000	1,750	>10,000

maximum degradation times from the sensitivity analysis presented in Appendix E of DOE Idaho (2003a). As indicated in Table 13, the failure times assumed for the PA are less than the failure times derived from the “base-case” degradation analysis and are less than or equal to the minimum times calculated from the sensitivity analysis.

4.2.6.3 Source Term Model

Radionuclide releases to the vadose zone were estimated by conducting one-dimensional transport simulations using the DUST–MS computer code (Sullivan, 2001). The two wasteforms for these DUST–MS simulations were the radionuclides in the grouted tank heel and piping and the radionuclides in the sand pad. The inventory used to develop the tank, piping, and sand pad source terms in the PA model (DOE Idaho, 2003a) is discussed in Sections 3.1 and 3.2. DUST–MS accounted for radionuclide partitioning between the infiltrating water and the wasteform (surface rinse model), as well as the retardation of radionuclide transport through the use of partition coefficients (K_ds) occurring in the grouted tank, sand pad, and vault floor.

4.2.6.4 Source Term Parameters

For the compliance case, radionuclide K_ds for the grout and concrete basemat were selected on the basis of literature reviews. Identical values for individual radionuclides were used for grout and concrete for strontium, iodine, and carbon. The values used in the compliance case for these three elements were identified as conservative and were selected by choosing the midpoint of the range of literature values (Portage, 2005d; DOE Idaho, 2006a). For strontium, literature values ranged from 0.001 to 0.006 m^3/kg, and DOE Idaho chose an intermediate value of 0.003 m^3/kg. For iodine, the literature range was 0.002 to 0.030 m^3/kg, and a value of 0.008 m^3/kg was used. For both strontium and iodine, the low end of the literature range corresponded to the recommended conservative value from the often-cited compendium by Bradbury and Sarott (1995). For carbon, DOE Idaho used a published recommended value of 5 m^3/kg, which lies in the middle of the range of cited literature values.

Different K_ds were used for technetium in the grout and in the concrete. The concrete value of 0.001 m^3/kg was taken from the Bradbury and Sarott (1995) recommendation for oxidizing conditions. For grout, DOE Idaho chose 2.5 m^3/kg, intermediate between the Bradbury and Sarott (1995) conservative value of 1 m^3/g for reducing conditionsh and a higher literature value of 5 m^3/kg. The selected technetium K_d for grout was much higher than the K_d for concrete because grout is assumed to provide a reducing environment for redox-sensitive technetium, whereas the concrete basemat is assumed to have an oxidizing environment.

For most elements, DOE Idaho (2003a) used different grout and concrete K_d values in their best-, realistic-, conservative- (coinciding with the compliance case), and worst-case scenarios (see Table 10b). The worst-case values tended to represent the lower bound of literature ranges, but also coincided with the recommended conservative values from Bradbury and Sarott (1995). These scenarios were used in sensitivity analyses to evaluate the effects of parameter selection on calculated dose.

For release from the contaminated sand pads, the K_d values used for strontium, iodine, carbon, and technetium were 0.015, 0.001, 0.005, and 0.0001 m^3/kg, respectively. These are sand soil values from the Sheppard and Thibault (1990) compendium. The technetium K_d selected for the sand pad was low, reflecting the assumed oxidizing condition in the sand pad. Strontium release from the sand pad appears to be most significant to dose (DOE Idaho, 2003a). Sensitivity analyses were not specifically conducted on the effects of sand pad K_ds on calculated release rates. DOE Idaho did implement an alternative transport model to simulate the initial contaminating event (see Section 3.1) and performed sensitivity analyses to quantify the uncertainty in the sand pad inventory (DOE Idaho, 2006a).

4.2.7 NRC Evaluation—Release and Engineered System Degradation

The engineered barrier system can be an important factor in mitigating the release and transport of radionuclides by limiting water contact with the waste and by retarding radionuclide transport to the vadose zone. The detailed analysis of degradation of engineered barriers (grouted tanks, vaults, and piping) conducted by DOE Idaho is based on reasonable conceptual models that consider the effects of important corrosion and chemical degradation mechanisms expected to occur at the INTEC TFF site. The failure times of the engineered components used in the performance assessment calculations are much shorter than the detailed analysis indicated. This approach of biasing engineered system failure toward pessimistic (i.e., earlier) values is adequate for the purpose of the DOE PA (2003a) modeling, considering the limited site-specific data (e.g., properties of the concrete and grout; chemistry of the soil moisture and water entering the vault) and the unvalidated degradation models.

4.2.7.1 Evaluation of Sorption Coefficients and Sorption Model

DOE Idaho choices for release (grout and sand pads) and engineered system transport (sand pads and concrete basemat) model K_d values were based on literature data; therefore, values selected for compliance demonstration should be defensibly conservative in the absence of site- and material-specific data. As discussed in more detail in clarifying RAI 17 (NRC, 2006a), this was not consistently the case. The conservative values used for the grout and concrete in the compliance case were typically chosen from the middle of a literature range without support for the specific value. In addition, values defined by DOE Idaho as worst case were typically equivalent to values defined in the literature as conservative. This was true in most cases for K_ds derived from the Bradbury and Sarott (1995) compendium for cementitious materials. (Note that for K_ds, the realistic and best scenarios were identical, and the label "realistic" was typically used for literature upper bound values, rather than being based on anything demonstrably realistic; see Table 10b.) Conservative K_d values, when used for a compliance demonstration, should be demonstrably bounding at the low end of reasonably expected values or the uncertainty of model results assessed and bounded. In many cases, DOE Idaho has used such lower bounds for worst-case values, with little explicit basis for conservative values.

DOE Idaho responded to an NRC staff comment on the choice of conservative K_ds by asserting that the conservative compliance case "is not analyzed as a bounding case, but as a 'reasonably conservative' case" (DOE Idaho, 2006a). DOE Idaho also referred to the sensitivity analysis results that showed that combining the worst-case partition coefficients (for both release and transport) with "conservative" inventory and infiltration parameters yielded modeled drinking water and all-pathways doses that were below performance objectives (DOE Idaho, 2006a,b). Although DOE Idaho did not conduct radionuclide release sensitivity analyses on sand pads K_ds, the chosen values appear reasonable.

The technetium K_d value selected for grout (2.5 m^3/kg) is potentially applicable to reducing conditions, but insufficient technical basis was provided by DOE Idaho for its assumption that the grout will provide a reducing environment. In response to NRC staff clarifying RAI 3 (NRC, 2006a) regarding this issue, DOE Idaho provided information on the components to be used in the grout mixture (DOE Idaho, 2006a). The components would include ground blast furnace slag, which is expected to ensure the establishment of a reducing environment that could mitigate technetium release. Although the amount of slag in the grout design may be sufficient to cause the formation of reducing conditions, steps must be taken by DOE Idaho to ensure the slag that is supplied by a vendor is reactive and will release its reducing agents. In particular, the slag should contain sufficient sulfur as sulfide to ensure reducing conditions will occur. Table 12 shows that the fully reducing K_d provides a significant barrier to the release of Tc-99 from the tank grout. More discussion on the importance of Tc-99 K_d on DOE Idaho's demonstration of compliance with 10 CFR Part 61, Subpart C, performance objectives is found in Section 4.4 and Appendix A.

The grout radionuclide release model relies on an assumption of equilibrium sorption of dissolved radionuclide to cementitious material. However, DOE Idaho has not demonstrated that a K_d approach is appropriate for this situation. Release from the grouted wasteform is a leaching process, and sorption is only one chemical process that may be affecting grout pore water radionuclide concentrations. Wasteform leaching experiments that are directly applicable to TFF grouted tank conditions would obviate the reliance on potentially conservative literature values in modeling release.

4.2.7.2 Evaluation of Alternative Conceptual Models for Release

DOE Idaho evaluated several alternative conceptual models for radionuclide release in its uncertainty analysis. For example, a quantitative analysis was conducted to compare the release rates using the DUST-MS surface rinse model to release rates using a first-order release rate model. Releases were computed assuming matrix diffusion into the shrinkage cracks between the grout and tank wall where fracture flow could occur. For both models, the waste was assumed to be present in the bottom 0.3 m [1 ft] of grout. Assuming the infiltration rate is high enough through the tank to cause fracture flow, peak release rates arising from matrix diffusion into shrinkage cracks are expected to be an order of magnitude smaller than the DUST-MS peak release rate. A sensitivity analysis was also conducted on the TFF tank releases for contaminants located at different locations within the DUST-MS model (Sullivan, 2001). This exercise evaluated the impact of incomplete mixing or layering of residual waste heels within the engineered grout pour on the results of the analysis. The residual tank contaminants were evaluated for three cases: (i) contaminants located in the grout 0.15 m [6 in] from the tank floor, (ii) contaminants located in the grout at the tank bottom, and (iii) contaminants located below the grout at the tank bottom (zero grout K_d effects). In addition, Tc-99 was modeled using K_ds representative of both reducing and oxidizing conditions. The oxidizing and reducing K_ds for I-129 and Sr-90 K_ds for the expected concrete degradation state for most of the 10,000 year simulation are the same (Bradbury and Sarrot, 1995) and therefore, no additional K_d analysis was performed for these two radionuclides. The worst-case configuration of tank waste led to a dose significantly higher than the PA results (about 20 times higher). The largest effect was on Tc-99 when the oxidizing K_d of 1 L/kg was used. However, the resultant dose was still under the 10 CFR 61.41 performance objective dose limit.

4.2.7.3 Summary

With respect to release and engineered system degradation, NRC staff concludes that sensitivity analyses show that DOE Idaho K_d choices alone will not affect conclusions regarding DOE Idaho's demonstration of compliance with performance objectives. However, as discussed in Sections 4.2.9 and 4.4, DOE Idaho will need to demonstrate that the grout formulation used in disposal actions will impose a robust reducing environment to manage uncertainty in the hydrogeologic model. Alternatively, DOE Idaho can reduce the uncertainty in the hydrogeologic model (see Sections 4.2.8 and 4.2.9). Based on the discussion above, recommendations for release modeling and parameter selection are discussed further in Appendix A.

4.2.8 Hydrology and Far-Field Transport

Given the depth to waste, the primary method of exposure of members of the public to residual radioactivity in the tanks and sand pads is through the groundwater pathway. After closure but prior to the end of active institutional controls, members of the public could potentially be exposed to residual radioactivity in the tanks, sand pads, and auxiliary equipment from releases into the vadose zone and transport through groundwater to the site boundary. Because the earliest releases from the grouted tank and auxiliary component systems are not assumed to occur until after 100 years, and the NRC staff agrees that this assumption is reasonable, calculating the dose to a member of the public during the institutional control period is moot. Exposures to members of the public after the end of active institutional controls, however, can occur through the same mechanism. DOE Order 435.1 defines the point of exposure for members of the public as either 100 m [300 ft] from the TFF or at the point of maximum exposure, as explained in Section 4.2.3.

Section 4.2.6 and 4.2.7 above discuss the mechanisms for concrete and grout degradation and near-field release of contaminants into the accessible environment. As discussed above, the concrete vault and grout between the vault wall and the tank wall are assumed to completely degrade after 100 years, at which time these system components are assumed to have the same hydraulic properties as the surrounding alluvium. Infiltrating water can, thus, interact with radionuclides in the contaminated sand pads and transport some amount of radioactivity to the hydrogeological environment. The tank wall is assumed to corrode, and the grout within the tank is assumed to completely degrade 500-years postclosure, at which time the grouted tank contents are assumed to have the same hydraulic properties as the surrounding alluvium. Again, infiltrating water contacting the tank residuals is assumed to transport the remaining radioactivity to the hydrogeological environment. Piping releases from degradation and corrosion are also assumed to occur after 500 years, but these releases are considered minor compared to the sand pads and tank residuals. As discussed above, the timing and magnitude of the release is dependent on the chemical properties of the grout, concrete, and sand pad. Although the system components are assumed to transmit water at the same rate as the surrounding media, the grout, concrete, and sand pads still provide a chemical barrier that limits releases of radioactivity into the environment. DOE Idaho used the PORFLOW variably saturated flow code (ACRI, 2000) to simulate two-dimensional water flow and radionuclide transport in the subsurface below the INTEC TFF following release.

Given limited site information, DOE Idaho made certain assumptions to estimate future hydrologic conditions relevant to the TFF and its inventory of radionuclides. Existing hydrologic and geologic data were used where possible. When data sources conflicted, DOE Idaho tried to confirm the data by consulting additional sources. If there was no information available, then DOE Idaho adopted an approach in which parameters were selected to produce the highest-possible transport rates. For example, if the only available data for a particular geologic unit was a range in hydraulic conductivity values, then the upper portion of the range was used to assign values to hydrostratigraphic units in the model.

4.2.8.1 Model Construction

The DOE PA hydrology model is based on a north-south oriented, 2,500 m [8,200 ft] USGS cross-section (DOE Idaho, 2003a) of the subsurface geology with a northern boundary at the BLR (Anderson, 1991). The cross-section contains alluvium, 19 basalt flow groups, and 11 continuous or discontinuous sedimentary interbeds (DOE Idaho, 2003a). With 103 vertical layers and 250 columns, DOE Idaho modeled a two-dimensional slice of the uppermost 200 m [660 ft] of the INTEC subsurface from the BLR in the north, through the center of two tank vaults, and southward (see Figure 17). The upper model boundary was located at the ground surface; the bottom model boundary was located 200 m (660 ft) below ground surface, consistent with the upper portion {i.e., upper 60 m [200 ft]} of the SRPA; and the location of the bottom boundary was selected to limit the amount of vertical dispersion that would affect dose calculations. The water table elevation varied between 134 and 139 m [440 and 456 ft] below the ground surface within the model domain. Vault floors at the TFF are approximately 14 m [45 ft] below ground surface and approximately 120 m [394 ft] above the SRPA. The model grid discretization varied, with a high density of columns at and adjacent to the two tanks and a high density of rows at and below the two tanks (see Figure 3-16 in the PA; DOE Idaho, 2003a).

4.2.8.2 Boundary Conditions

Uniform net infiltration fluxes are applied as an upper boundary condition. Two grid blocks in the uppermost northern edge of the model domain were injected with water at a combined rate of 6.98 m^3/d [247 ft^3/d] to simulate BLR seepage based on historical gauging data presented by Bennett (1990). Infiltration rates reported in the literature at INL at the time of model construction ranged from 0.41 to 12.0 cm/yr [0.16 to 4.9 in/yr] (DOE Idaho, 2003a; Parker, 2006). The infiltration rate was considered an uncertain model parameter and included as part of the sensitivity analyses. Interim infiltration control actions to reduce the infiltration through the TFF soils (see Sections 4.2.4 and 4.2.5) were not simulated in the PA modeling. A relatively low infiltration rate of 4.1 cm/yr [1.6 in/yr] was used for the compliance analysis to simulate expected infiltration through the earthen cover, and the highest infiltration rate used in the uncertainty analysis was 12 cm/yr [4.7 in/yr]. The north and south vadose zone boundaries were no-flow, and the north and south saturated zone boundaries were set to constant hydraulic head values based on the regional potentiometric surface. A no-flow boundary condition was used for the lower boundary condition. Model grid blocks that represent the SRPA were uniformly assigned a single property set. Model grid blocks that represent vadose geologic units, however, were assigned their own individual property sets. Overall, 20 separate subhorizontal zones were assigned a hydraulic conductivity value different from the value assigned to a vertically adjacent zone.

4.2.8.3 Material Properties

The hydraulic conductivity of alluvial grid blocks was set at 80 m/d (300 ft/d), a value consistent with that reported by Freeze and Cherry (1979) for a coarse sand and gravel lithology. Alluvial porosity was based on values presented by Magnuson (1995), and the air-water capillary pressure function of water saturation was adapted from a publication by Blumb, et al. (1992). The curve for the vadose hydraulic conductivity of alluvium as a function of water saturation was developed using the methodology presented by van Genuchten (1978).

Hydrologic property data for unsaturated basalt flow groups at INL are limited. Anderson, et al. (1999) assessed the relationship between the thickness of basalt layers and hydraulic conductivity. They concluded that stratigraphic intervals comprising thin basalt layers have more contacts, rubble zones, and cooling fractures. Hence, DOE Idaho modeled (DOE Idaho, 2003a) thin stratigraphic intervals with higher hydraulic conductivity and thick stratigraphic intervals with lower hydraulic conductivity. In response to NRC's RAI (NRC, 2006a), DOE Idaho stated that the final calibrated basalt units comprising thick flows were assigned a vertical hydraulic conductivity value of 10 m/d [33 ft/d] (DOE Idaho, 2006a). Basalt units comprising thin flows were assigned a vertical hydraulic conductivity value of 100 m/d (330 ft/d). Each basalt unit was assigned a horizontal hydraulic conductivity of 1 m/d [3.3 ft/d] (DOE Idaho, 2006a). In response to NRC's clarifying RAI 7 (NRC, 2006a), DOE Idaho clarified that the constitutive relationships in Figure 2-20 of the PA (DOE Idaho, 2003a) were used for the basalts in the groundwater analyses (DOE Idaho, 2006a). DOE Idaho used an approach that results in rapid water flow through the fractures (simulated as porous media) consistent with a conceptual model where water released from sedimentary interbeds moves rapidly through the basalts with little or no residual water left in the fractures. DOE Idaho characterized this approach as conservative, citing recent work by Magnuson (2004) that showed moisture retention curves similar to those used in the PA resulted in a conservative approach to simulating flow in the vadose zone (DOE Idaho, 2006a).

4.2.8.4 Calibration to Perched Zones

Perched water bodies beneath the TFF receive seepage from the BLR, but most of the modeled perched zones increased in thickness and lateral extent when percolation ponds located immediately south of INTEC were simulated by DOE Idaho. These percolation ponds were decommissioned, as discussed in Section 4.2.4. The factors that control the perched water bodies are not well understood. DOE Idaho postulates that the lithologic features that contribute to vertical hydraulic conductivity contrasts between the basalt layers and sedimentary interbeds provide the mechanisms for the development of perched water bodies. Assumptions related to the perched water bodies are very important to numerical model development. DOE Idaho used knowledge of the location and extent of perched water bodies to calibrate their flow model, and they made the explicit assumption that the occurrence of perched water is due to the presence of permeability barriers (e.g., low permeability sedimentary interbeds). DOE Idaho developed a calibrated model using existing observations such as the distribution and extent of perched water bodies (DOE Idaho, 2003a). A difficulty is that the major source of water for the perched water bodies is man-made. An anthropogenic source for the southern perched water bodies no longer exists due to the percolation pond closure. Additionally, there is uncertainty regarding the exact locations of the perched zones because of the limited number of wells available for calibration.

DOE used an incremental modeling process to assess the impacts of key input parameters on numerical simulations. The hydraulic conductivities of unsaturated sedimentary interbeds were established by an iterative process that required model-based matching of the known extent of perched water (see Figures 3-17 and 3-18; DOE Idaho, 2003a), given data on the volumes of present-day infiltration water. DOE provided the resulting horizontal and vertical hydraulic conductivities for individual sedimentary interbeds in response to NRC staff's RAI in

Table CR–8–1 (DOE Idaho, 2006a). The calibrated conductivity was within the measured range for these lithologies (DOE Idaho, 2003a). Sensitivity analyses conducted by DOE indicate the model is sensitive to variations in vertical hydraulic conductivity for the key hydrostratigraphic units.

4.2.8.5 Calibration to Analytical Data

DOE Idaho conducted a transport verification simulation that was based on actual tritium discharges to the percolation ponds during the period between 1984 and 1988. Tritium concentrations predicted by the model were higher than, yet comparable to, field measurements (DOE Idaho, 2003a; Cecil, et al., 1991). In response to NRC's clarifying RAI 16 (NRC, 2006a), DOE Idaho provided chloride analytical data from monitoring wells in the vicinity of the percolation ponds for comparison (DOE Idaho, 2006a).

4.2.8.6 Integration of DUST-MS Release Model with PORFLOW Transport Model

The calibrated hydrology model defined the steady-state velocity vector field for the transport model. Transport from the facility was simulated for a period of 1 million years. Radionuclide release rates predicted by DUST–MS were input to PORFLOW as time history fluxes beginning at the time of engineered barrier degradation (i.e., when the hydraulic conductivities of the engineered barrier system change from impermeable values to degraded values). In response to NRC staff's clarifying RAI 15 (NRC, 2006a) regarding integration of the DUST-MS and PORFLOW models, DOE Idaho stated that the DUST-MS release rates were adjusted by multiplying the ratio of the two-dimensional slice area for one tank used in PORFLOW to the area of an entire tank used in DUST-MS (i.e., 0.084) for input into the PORFLOW model. This adjusted release rate was input by DOE Idaho into PORFLOW at the grid nodes representing each tank location (i.e., two tanks in the model).[14] Transport simulations were conducted separately for individual sources (i.e., piping, sand pads, and tanks) to assess their singular impacts. Finally, the groundwater concentrations from piping, sand pad, and tank releases were summed to obtain the total concentration of a radionuclide in groundwater at the point of maximum exposure.

4.2.8.7 Model Results that Determine Compliance Point

Modeled radionuclide concentrations were observed to enter the SRPA substantially further from the TFF than the typical 100-m [330-ft] compliance point. According to the DOE PA, the model exhibits deflection of contamination by low permeability sedimentary beds (DOE Idaho, 2003a). Modeled permeability barriers impede the direct vertical transport of radionuclides. As assumed by DOE Idaho, the perched water zones from BLR seepage are water sources for the surrounding vadose zone. According to the DOE PA (2003a), capillary and gravitational forces (total hydraulic head) redirect the migration path of radionuclides around the upper perched zone. DOE Idaho stated in its PA (DOE Idaho, 2003a) that some fraction of radionuclides enter perched zones, but the majority is transported around perched zones (see Figure 18).[15]

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[14] A portion of the response to clarifying RAI 15 is inconsistent with the response to clarifying RAI 19, which states that only one source-loading location, the grid location of the southernmost tank, was modeled in the analysis, and the resulting receptor concentrations were doubled.

[15] It is important to note that the figure DOE Idaho provided to NRC to correct the locations of sedimentary interbeds resulted in inconsistent display of the center-line of the Tc-99 plume (see Figure 18) that shows a center-line of 100 pCi/L in DOE Idaho (2006c), compared to Figure 4-2 in DOE Idaho (2003a) that shows a center-line of

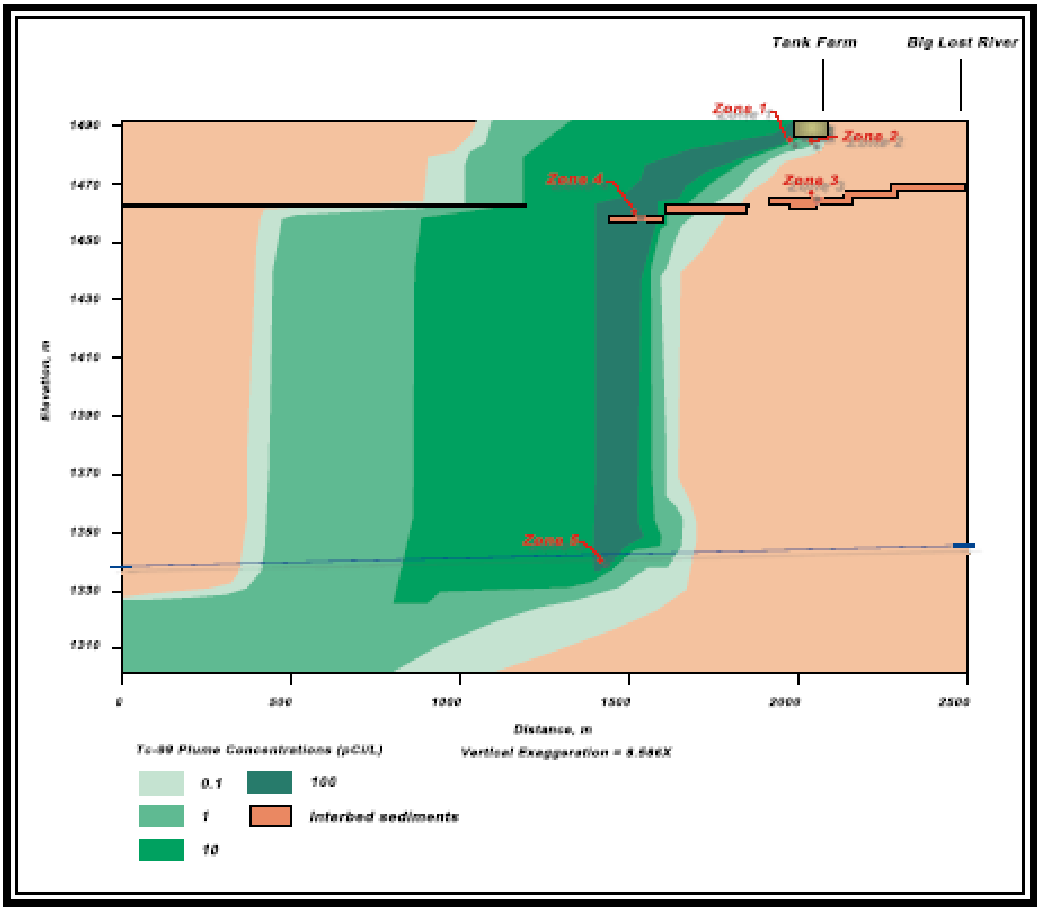

Figure 18. Cross Section of Tc-99 Plume (From DOE Idaho, 2006c)

Locations where interbeds are thin to absent greatly affect the downward migration of radionuclides. Consequently, DOE Idaho concluded that the hydrostratigraphy defined by the Anderson, et al. (1991) and representation methods implemented in the model (DOE Idaho, 2003a) both played a major role in the modeled migration pattern. DOE Idaho concluded that the 100-m [330-ft] downgradient compliance point was an inappropriate location at which to quantify the groundwater pathway dose to a member of the public because the maximum modeled concentrations occur approximately 600 m [2,000 ft] downgradient from the tanks. Thus, this point of maximum concentration in groundwater was chosen for the compliance point, and it is at this location that nodal contaminant concentrations were averaged for the upper 10 m [33 ft] of the aquifer to estimate the concentrations expected in a water well screen of this length.

1000 pCi/L. Based on the groundwater concentrations presented in the PA, it appears the figures presented in the PA and RAI responses are old figures from previous modeling when much higher release rates for Tc-99 were predicted with DUST-MS. In response to clarifying RAI 14 (NRC, 2006a), DOE Idaho stated (DOE Idaho, 2006a) that it retained the old modeling results for release rates in Appendix F to show the evolution of the PA modeling. Appendix F in the PA also has the screening analyses; therefore, it appears that this information is valuable, although confusion results when DOE Idaho presents old information as if it were part of the new analysis without clarification.

4.2.8.8 Transport Parameters

DOE Idaho selected longitudinal and transverse dispersivity values for the vadose zone based on a literature review. These values were constrained by a simple heuristic that is based on the scale of transport through individual hydrologic units (DOE Idaho, 2003a). Longitudinal dispersivity values of 0.3 m [1.0 ft] for interbeds and 1.9 m [6.1 ft] for basalts were assigned for transport simulations. Transverse dispersivities were set at one-half the values of the longitudinal dispersivities. Dispersion was modeled to occur only in the longitudinal and vertically transverse directions. DOE Idaho states that this is a conservative approach for predicting downgradient contaminant concentrations because horizontally transverse dispersion is neglected.

For the transport model, sorption coefficients were assigned to the modeled radionuclides (C-14, Sr-90, Tc-99, and I-129) on the basis of a survey of the literature and limited site-specific information (DOE Idaho, 2003a; Portage, 2005d). Separate sorption coefficients were developed for the basalts, which constitute the majority of the subsurface strata, and also for sedimentary interbeds in the basalt sequences. DOE Idaho (2006a) justified assuming that fractured basalt would be effective at retarding radionuclide migration by referring to the presence of sorptive iron oxides on fracture surfaces, citing experiments on crushed basalt, and supporting sensitivity analyses. DOE Idaho performed sensitivity analyses to investigate the impact of transport parameters, infiltration rate, and inventory on the model results (see Tables 10a and 10b). The best and realistic values were identical for most parameters. Therefore, DOE Idaho generally used three scenarios (realistic and best; conservative; and worst case) to propagate uncertainty in the PA analyses (DOE Idaho, 2003a), thereby reducing the number of combinations requiring evaluation. DOE Idaho used K_d values deemed conservative for the compliance case. For interbedded sediments, these were 0.01 m^3/kg for carbon, 0.0001 m^3/kg for iodine, 0.018 m^3/kg for strontium, and 0.00001 m^3/kg for technetium. The basalt values were 0.005 m^3/kg for carbon, 0.0001 m^3/kg for iodine, 0.006 m^3/kg for strontium, and 0.00001 m^3/kg for technetium. Of these four elements, DOE Idaho cited site data only for strontium and technetium (Portage, 2005d); the site data were used only for strontium. For the compliance case, DOE Idaho tended to select intermediate values from site or literature ranges. The worst-case value was chosen as a lower bound to available data. In the sensitivity analyses, when worst-case K_ds were used in combination with conservative inventory and infiltration parameters, modeled drinking water and all-pathways doses were below performance objectives (DOE Idaho, 2006a, 2003b). However, when worst-case infiltration rates and transport parameters were used, the dose was predicted to be 85 mrem/yr, above the performance objective limit of 25 mrem/yr. Because the worst-case peak dose is a result of Sr-90 released from the sand pad and sensitivity of model results to sand pad inventory was not investigated in the PA (DOE Idaho, 2003a), the worst-case dose predicted by DOE Idaho is same for all inventory cases.

4.2.8.9 Flooding Flow and Transport Simulation

The PMF represents the hypothetical flood considered the most severe flood event reasonably possible based on hydrometeorological application of maximum precipitation and other hydrologic factors. The probable maximum flood is assumed to result from an overtopping failure of the 24-m [79 ft]-high earth-filled Mackay Dam caused by a general storm probable maximum precipitation (PMP) event (Idaho National Engineering and Environmental Laboratory, 1999; Koslow and van Haaften, 1986). The inundation map from this probable maximum flood was given in Figure 2-18 of the PA (DOE Idaho, 2003a) and in higher resolution (see Figure 19) in DOE Idaho’s response (DOE Idaho, 2006a) to NRC staff’s RAIs (NRC, 2006a). The resulting

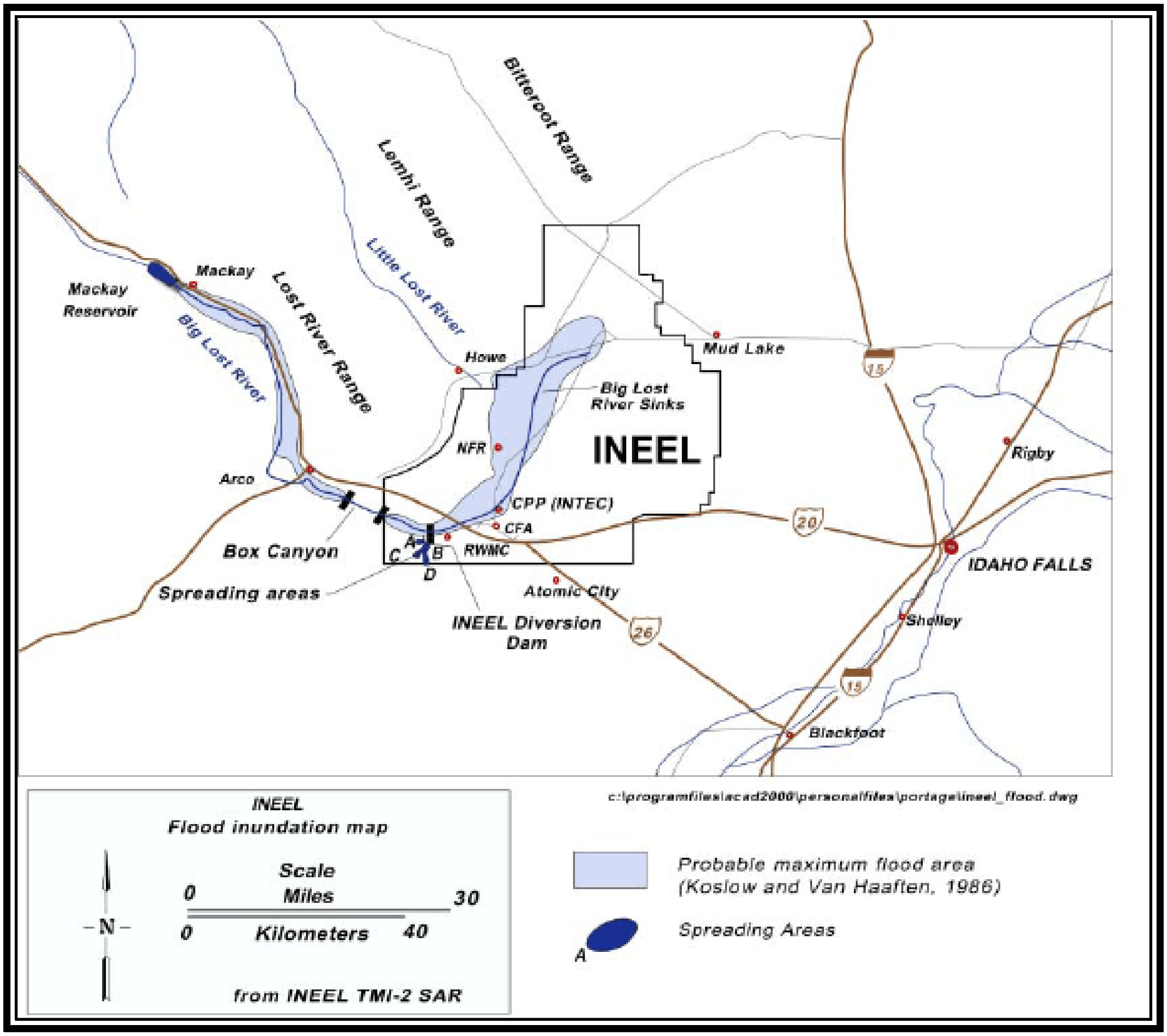

Figure 19. Location of INEEL Diversion Dam and Mackay Dam

peak flow from the probable maximum precipitation-induced dam failure is 8,685 m^3/s [306,700 cfs] in the reach immediately downstream of the Mackay Dam, approximately 2,035 m^3/s [71,850 cfs] at the INL Diversion Dam, and 1,892 m^3/s [66,830 cfs] at INTEC. The flood wave is expected to reach INTEC in 13.5 hours after dam failure. Flood water velocities are estimated to range from 0.3 to 0.9 m/s [1 to 3 ft/s] near the Flood Diversion Facility, and the model result for peak water velocity at INTEC is 0.8 m/s [2.7 ft/s] (Koslow and van Haaften, 1986). The TFF site elevation is approximately 0 to 1 m [0 to 3 ft] below the estimated maximum elevation of floodwater {i.e., 1,498 m [4,917 ft]} for this scenario, with the majority of tanks located where the surface elevation is 0.3 m [1 ft] below the floodwater elevation (DOE Idaho, 2003a). The INL Flood Diversion Facility may divert some floodwater if the Mackay Dam fails, assuming it does not lose significant capacity to sedimentation after the period of institutional control.

The DOE Idaho PA discusses the potential for flooding at INTEC as a result of seismically induced dam failure. The Mackay Dam was classified as a high-hazard dam by the State of

Idaho in a 1978 inspection that used the U.S. Army Corps of Engineers guidelines for inspecting dams for safety. The Mackay Dam is in a region where large earthquakes have occurred in the past, including the 1983 Borah Peak earthquake. The Mackay Dam was not damaged by this earthquake, demonstrating stability of the embankment during moderate vibratory ground motions. As noted in the DOE Idaho PA, however, the Mackay Dam was built without seismic design criteria, which led DOE Idaho to conduct analyses of potential flooding impacts at the site in the event of seismically induced dam failure (Koslow and van Haaften, 1986). This scenario is less likely than overtopping failure and thus would have less serious impact at INTEC than does the probable maximum precipitation-induced probable maximum flood.

The impact of a flood resulting from failure of the Mackay Dam on radionuclide transport was analyzed by DOE. The flood was assumed to occur at the time of tank failure (500 years after closure). Infiltration was increased to 100 times the 12.4 cm/yr (4.9 in/yr) worst-case scenario infiltration rate. The DOE Idaho PA results suggested that the transport simulations were very sensitive to infiltration rate because they not only affected the transport rate through the vadose zone, but also the release rate from the wasteform (DOE Idaho, 2003a). While the modeled peak concentrations at the water table were lower for the flooding scenario than for the conservative or compliance case (non-flooding) scenario due to dilution, the modeled postpeak concentrations at the water table were higher for the flooding scenario than for DOE Idaho's conservative or compliance case scenario because the floodwater pulse mobilized more radionuclides more quickly than would be mobilized in the absence of a flood. Approximately 30 years after initial mobilization from floodwater, dilution effects become negligible, and radionuclides mobilized during the flooding episode make a more significant contribution to dose than observed in DOE Idaho's conservative or compliance case scenario (see Figure 7-15; DOE Idaho, 2003a); however, dose estimates still remain below the performance objectives.

4.2.9 NRC Evaluation—Hydrology and Far-Field Transport

DOE Idaho attempted to manage uncertain hydrologic and geologic information by adopting an approach whereby the values chosen for the hydrologic transport parameters yielded reasonably high transport rates. However, conceptual model uncertainty can significantly impact risk estimates, especially if the flow and transport occurs through a complex hydrogeological system. For INL, the fractured, discontinuously interbedded, unsaturated hydrologic system poses many challenges for effectively modeling flow and transport for the purpose of making long-term predictions for assessment analyses. In fact, based on recent monitoring data, subsequent characterization activities, and updated modeling, the HCM for TFF has recently evolved. NRC staff considered ongoing CERCLA characterization and documentation to risk inform the review (e.g., additional and more recent data analysis and modeling was used to evaluate DOE's PA model and demonstration of compliance with 10 CFR Part 61 performance objectives).

4.2.9.1 Evaluation—Conceptual Model Development, Model Construction, and Model Support

The selected USGS cross section used to construct the PA model runs from north to south across INTEC, consistent with the expected regional groundwater flow direction. However, the local direction of vadose zone flow underneath the TFF is not necessarily consistent with the model orientation. The BLR channel trends approximately 31 degrees north of east at the location where its recharge most directly affects flow in the vadose zone below the TFF. Recharge waters will tend to flow in the vadose zone vertically down and laterally away (in a southeasterly direction within the model domain) from the BLR channel. At INTEC, perched water contours below the TFF show a propensity to shed water to the southeast (see Figure 2-20; Rodriguez, 1997). Because the stream channel strikes southwest-northeast, perched water located southeast of the stream channel is also anticipated to pond with a predominantly southwest-northeast strike. As a result, the affected sedimentary interbeds underneath the TFF are anticipated to shed naturally occurring perched water in a southeasterly direction. The orientation of the model and discretization of sedimentary interbeds (discussed further below) may affect the flow paths, distances, and thus, the travel time of HRRs. DOE Idaho PA model construction (DOE Idaho, 2003a) may also help explain potential differences in model-predicted flow directions and distances versus those that can be inferred from monitoring well data (DOE Idaho, 2006d).

The location of a newly installed monitoring well (ICPP-MON-A-230 located north of the TFF, see Figure 7) where elevated Tc-99 groundwater concentrations were detected, suggested that Tc-99 contamination linked to a TFF piping release (see CPP-31 release site on Figure 6) may have entered the SRPA significantly closer to the TFF (DOE Idaho, 2004) than in DOE PA model predictions (DOE Idaho, 2003a) that show entry of contaminants 600 m (1800 ft) south of the TFF near well USGS-48 (see Figure 7). Additionally, a newly constructed well located 1,500 ft from ICPP-MON-A-230 also indicates that the extent of the Tc-99 plume is more widespread than originally thought (DOE Idaho, 2006e).

There is also some uncertainty with respect to the extent to which the BLR affects the perched zone. More recent RI/BRA modeling suggests that the BLR has minimal impact on the perched zone, as evidenced by the lack of response in wells screened in the upper perched zone following flow in the BLR in 2005 (DOE Idaho, 2006e). Furthermore, the BLR did not flow from 2000 to 2005, yet the perched zone persisted during this time period, suggesting that other sources (e.g., precipitation infiltration and service water leakage) are responsible for the persistence of the northern perched zone (DOE Idaho, 2006e). The DOE PA (DOE Idaho, 2003a) suggests that perched water causes lateral spread of the plume in the final calibrated model. Thus, the influence of BLR seepage on the creation of the perched zones is emphasized in the PA. However, DOE Idaho provided a cross-section (DOE Idaho, 2006c, see Figure 20) of the final calibrated model in response to an NRC information request (NRC, 2006c), which shows a small areal extent of the perched water close to the BLR (within a few hundred feet) above the upper sedimentary interbed. The perched zone does not extend continuously and laterally to the “spillway” 600–700 m (1,970–2300 ft) away from the TFF where the modeled Tc-99 plume shows vertical transport toward the aquifer (see Figure 18) where a receptor could be exposed. Thus, contaminants are not deflected by the perched zone to the modeled “spillway” contrary to explanations in the DOE PA documentation (DOE Idaho, 2003a). DOE Idaho provided additional information (DOE Idaho, 2006c) regarding the large lateral extent of the PA modeled contaminant plume (DOE Idaho, 2003a, see Figure 18) in response (DOE Idaho, 2006c) to an action item (NRC, 2006c) from the June 1, 2006, public meeting. That information indicates that the pressure gradient caused by the BLR boundary condition in the model is actually responsible for the lateral spread of the plume. DOE Idaho’s recent clarification (DOE Idaho, 2006c) is consistent with the DOE PA model results (DOE Idaho, 2003a).

The uncertainty in BLR seepage was not investigated in the uncertainty and sensitivity analysis. The PA model (DOE Idaho, 2003a) does not consider the transient nature of BLR seepage and may likely overestimate the impact of BLR seepage on flow and transport at the TFF (e.g, only a fraction of the BLR seepage should have been used as a boundary condition because the model domain represents only the southern portion of the site receiving BLR seepage in the vicinity of the TFF). In the most recent RI/BRA modeling for the TFF (DOE Idaho, 2006e), the

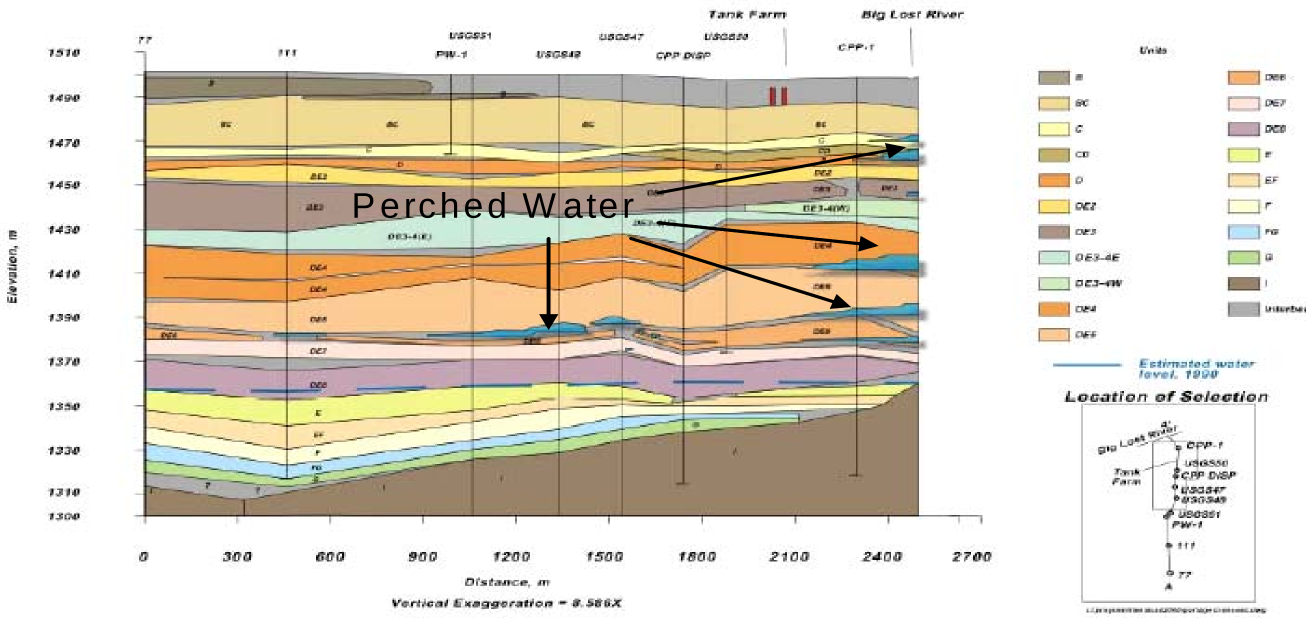

Figure 20. Cross Section of Final Calibrated Model (From DOE Idaho, 2006c)

transient seepage rates used for the BLR ranged from 0–3 m^3/d [0–106 ft^3/d], considerably lower than the 7 m^3/d [247 ft^3/d] assumed by DOE Idaho in its evaluation under the premise that higher seepage rates would be conservative. Considering the effect BLR seepage has on the flow field at TFF in the PA model (see Figure 18), the conservatism of this assumption was not fully supported.

Discretization and parameterization of hydrostratigraphic units in the DOE PA model (DOE Idaho, 2003a) may also significantly impact predicted flow paths, transport distances, and resultant model predictions. NRC staff has noted the following potential limitations with the PORFLOW model:

- Delineation and material property assignment of geologic features present at INTEC such as volcanic vents, dikes, and basaltic rubble zones may have a significant impact on contaminant flow and transport, e.g., rubble zones could quickly transport radionuclides to an interbed discontinuity located much closer to the TFF.

- Disturbed alluvium (from operations at INTEC) and historical BLR channel deposits may have much higher hydraulic conductivities than represented in the modeling.

- Unrealistic discretization, both horizontally and vertically, of discontinuous sedimentary interbeds may have a significant effect on travel distance and travel time to the saturated zone (e.g., combining discontinuous interbeds into longer, thicker, or continuous interbeds can affect travel distances and time, and therefore sorption and decay along the flow path).

The uncertainties associated with DOE Idaho's hydrologic modeling identified above have the most impact on Sr-90 contaminant flow and transport. Sr-90 has a relatively short half-life

(30 years) relative to DOE Idaho PA (DOE Idaho, 2003a) model-predicted transport time to the aquifer, approximately 30 yr for non-sorbing constituents (approximately 550 years for sorbing Sr-90 due to attenuation of Sr-90 in the sand pad, concrete vault, and 600 m [2,000 ft] of lateral transport path in the vadose zone). The uncertainty identified by NRC staff in Sr-90 contaminant transport is addressed by multiple lines of evidence that consider the likely bounding inventory for Sr-90 in the sand pad (Section 3.2); the likely pessimistic performance assumed in the PA model for the grouted vault (Section 4.2.6 and 4.2.7); and observations of contaminant flow and transport from monitoring data related to historical releases that provide a basis for the expected attenuation of Sr-90 in the subsurface at INTEC. Additional support for PA model predictions (DOE Idaho, 2003a) is discussed further below.

4.2.9.2 Evaluation—Transport Parameters

Dispersion of contaminants was only assumed to occur in the longitudinal and vertically transverse directions. While DOE Idaho suggests this is a conservative approach to the prediction of downgradient contaminant concentrations because horizontally transverse dispersion (dispersion in the direction perpendicular to the two-dimensional model) is neglected, NRC staff does not agree that this treatment is strictly conservative, because the modeled tanks are located in-between two other tanks in the horizontally transverse direction. Any loss of mass due to dispersion in the horizontally transverse direction away from one tank can be assumed to be gained through dispersion from an adjacent tank. Only the tanks on the periphery and contamination from the sand pads would tend to lose mass in the horizontally transverse direction without a comparable gain.

Regarding sorption parameters, the interpretation of the results of flow and transport models for a site-specific application can be difficult when the models rely heavily on generic information. DOE Idaho’s choices for natural system transport model K_d values for carbon, iodine, and technetium were based on literature data; therefore, values selected for compliance demonstration should be defensibly conservative in the absence of site- and material-specific data. As discussed in more detail by NRC (2006a, clarifying RAI 17), this was not consistently the case. Conservative values used for the grout and concrete in the compliance case were typically chosen from the middle of a literature range without support for the specific value. In addition, values defined by DOE Idaho as worst case were typically equivalent to values defined in the literature as conservative.

Considerably more site-specific data exist on Sr-90 sorption at INL (Portage, 2005d; DOE Idaho, 2006a; Porro, et al., 2000). The selected basalt compliance K_d of 0.006 m^3/kg lies within the range of reported values (0.0011 to 0.029 m^3/kg). However, the Sr-90 conservative or compliance case value is well above reported site-specific values as low as 0.0011 to 0.0027 m^3/kg (Del Debbio and Thomas, 1989) and 0.0025 to 0.0043 m^3/kg (Porro, et al., 2000). DOE Idaho supported this choice by pointing to the entire range of reported values of 0.0011 to 0.029 m^3/kg. However, in the report from which the highest values were published (Colello, et al., 1998), the majority of K_ds are < 0.010 m^3/kg, and DOE Idaho performed no detailed critical review of the applicability of the higher values (DOE Idaho, 2006b). Furthermore, a realistic value of 0.013 m^3/kg seems high considering that the majority of measured values are significantly lower. The selected strontium K_d value for sedimentary interbeds was 0.018 m^3/kg, which is reasonable in light of the generally higher values reported for INL sediments (DOE Idaho, 2006a). Sr-90 sorption may be affected by the large historical release of radiological contaminants to the shallower alluvium located above the TFF, although this effect is expected to be minimal over longer time (release occurred in 1972) and distance (release occurred in the shallow alluvium) scales. The effects of high-ionic strength SBW waste leakage and competitive sorption on Sr-90 transport in the alluvium were evaluated in the updated RI/BRA modeling through additional geochemical modeling (DOE Idaho, 2006e). The results of this modeling show that while variable sorption modeling was necessary during transport of the SBW plume through the alluvium, current sorption is expected to be near steady-state and a constant K_d can be used for future transport modeling of Sr-90 through the alluvium. Most of the future impact on the SRPA from Sr-90 will be from the contaminated sediments in the perched water below the TFF.

DOE Idaho responded to an NRC staff comment on the choice of conservative K_ds by asserting that the conservative compliance case "is not analyzed as a bounding case, but as a reasonably conservative case" (DOE Idaho, 2006a). DOE Idaho also referred to the sensitivity analysis results that showed that combining the worst-case partition coefficients (for both release and transport) with conservative inventory for the tanks and the conservative infiltration rate yielded an all-pathways dose of 0.15 mSv/yr [15 mrem/yr] that is significantly below the performance objective (DOE Idaho, 2006a, 2003a). However, as discussed in Section 4.2.5 above, the infiltration rate selected by DOE Idaho is not considered conservative with no infiltration controls in place. DOE Idaho used a "conservative" infiltration rate of 4 cm/yr [1.6 in/yr], but the latest modeling report for TFF used 18 cm/yr [7 in/yr] based on additional data collection and analysis (DOE Idaho, 2006e). Table 10a sensitivity analysis results for the worst-case transport simulations and "high" infiltration rate (12 cm/yr) show that Sr-90 would exceed the performance objective (0.85 mSv/yr (85 mrem/yr) all-pathways dose compared to 0.25 mSv/yr [25 mrem/yr] limit) for any inventory (because the Sr-90 sand pad inventory is not varied). It is important to note that the same sand pad K_d was used for all sensitivity simulations and therefore, the worst-case transport parameters essentially only evaluate the worst-case vadose zone parameters. The uncertainty in the sand pad inventory for Sr-90 is mitigated by indirect vault liquid sampling data that provides support that the Sr-90 estimated inventory for the sand pad is bounding (see Section 3.2). Additionally, observations regarding the potential attenuation of Sr-90 in perched water and the SRPA below the INTEC can be used to constrain model predictions as discussed below. The uncertainty in Sr-90 transport in the vadose zone at INL is also mitigated by the likely conservative assumption regarding the release of Sr-90 from the sand pad at 100 years. DOE Idaho provided additional information that shows that the release of Sr-90 from the sand pad will be significantly reduced if the grouted vault is not assumed to fail until 250 years postclosure (a reduction factor of about 300 compared to about 12 after 100 years when the vault and vault grout is assumed to fail), making the doses from Sr-90 insignificant as more optimistic assumptions are made regarding the performance of the grout (DOE Idaho, 2006c). Table 14 shows the contributions that engineered and natural barriers make towards reducing predicted contaminant concentrations to levels below performance objectives in the SRPA where a receptor could be exposed.

Because credit is taken for Sr-90 attenuation in the sand and concrete floor which provide a significant chemical barrier to the release of Sr-90 out of the vault for more than an additional 100 years after the assumed vault failure, it is still important to manage the uncertainty in Sr-90 release and transport with multiple lines of evidence. DOE Idaho also took credit for Tc-99 sorption in the reducing grout in the tank (see Tables 12 or 14). In the case of Tc-99, the tank grout is expected to provide a more significant barrier to the release of radioactivity into the environment (e.g., 99.99 percent fraction sorbed for Tc-99 in the reducing tank grout in the compliance case).

Hydraulic dispersivity, distribution coefficients (K_ds), and net infiltration rates were subject to a sensitivity and uncertainty analysis, as recommended by NRC staff (Essig, 2002). Results indicate that dose estimates are more sensitive to changes in transport parameters or infiltration

Table 14. Summary of NRC Staff Perspective on Credit for Engineered and Natural Barrier Performance*			
	Tc-99	**Sr-90**	**I-129**
Minimum Total Barrier Performance Needed for Compliance†	4 orders of magnitude	9 orders of magnitude	3 orders of magnitude
Engineered Barrier (most effective of grouted tank, vault, or sand pad)	1 to 4 orders of magnitude	4 orders of magnitude‡	1 to 2 orders of magnitude
Natural System (Unsaturated Zone)		3 to 4 orders of magnitude§	
Natural System (Saturated Zone)‖	1 to 2 orders of magnitude	1 to 2 orders of magnitude	1 to 2 orders of magnitude

*This table presents NRC staff's perspective on the credit DOE Idaho can reasonably take for engineered and natural system performance in attenuating releases of Tc-99, Sr-90, and I-129, from the INTEC TFF, given the limitations in its groundwater model. Similar to Table 12, row 1 (highlighted in grey) provides a rough factor (within an order of magnitude) reduction necessary in the waste pore water concentration to achieve levels that will meet the 10 CFR 61.41 dose-based performance objective of 25 mrem/yr. Rows 2 through 4 provide a rough factor reduction in concentration and dose attributable to various barriers as indicated. Natural system performance (Rows 3 and 4) is calculated by NRC and is broken down into two components–unsaturated zone and saturated zone attenuation–this differs from Table 12 which presents DOE Idaho's credit for natural system performance based on its performance assessment modeling.

†Row 1 is based on the maximum, possible pore water concentration. The concentration used for this calculation is recognized as being very pessimistic because actual exposure to a receptor at the maximum concentration is virtually impossible.

‡The Sr-90 dose can be completely eliminated with more optimistic assumptions regarding barrier performance (e.g., if the tank vaults remain intact for a few hundred years, the short-lived radionuclides will decay to negligible levels).

§The factor reduction for Sr-90 represents NRC staff's perspective on an expected average (considers potential flow paths that may by-pass sedimentary interbeds) attenuation of Sr-90 in the subsurface from contact of contamination with basalts and sedimentary interbeds. This factor represents a reasonably conservative estimate based on NRC staff calculations and observed attenuation of SBW from historical releases.

‖Row 4 natural system concentration reduction factors for the saturated zone are based on NRC staff calculations of the expected dilution in the SRPA.

rates than they are to changes in source inventory given the range in values. Additionally, dose and arrival times are more sensitive to variations in infiltration rate and lithologic distribution coefficients than they are to variations in hydraulic dispersivity values. Finally, dose estimates for certain radionuclides (e.g., Tc-99) are affected more by the choice of grout distribution coefficients than by the choice of lithologic distribution coefficients (see Tables 10a, 12, and 14). The impact of hydrologic model construction uncertainty is expected to be more severe for Sr-90 than for Tc-99 and I-129, which are relatively non-sorbing (in the vadose zone). This is consistent with the RI/BRA modeling that shows Sr-90 concentrations are extremely sensitive to the assumed adsorptive capacity of the interbeds that strongly affect travel time and decay (DOE Idaho, 2006e).

4.2.9.3 Evaluation—Model Calibration

As discussed above, in response to NRC staff's clarifying RAI 16 (NRC, 2006a), DOE Idaho provided (DOE Idaho, 2006a) chloride analytical data from monitoring wells in the vicinity of the percolation ponds to show that the flow model was well calibrated (chloride is a conservative tracer). However, a closer comparison of the data provided by DOE Idaho reveals that the spatial distribution of contamination predicted by the PA model suggests more contamination flows north of the decommissioned percolation ponds, whereas the site data suggest the contamination primarily flows to the south. Therefore, the tracer data does not support the flow direction predicted by the PA model.

The final calibrated model (DOE Idaho, 2003a, see Figure 20) results are inconsistent with the current extent of the perched zone based on recent monitoring data (DOE Idaho, 2006d). The northern, shallow perched zone has persisted even in the absence of the percolation and wastewater treatment seepage ponds (see Section 4.2.4). Thus, the final calibrated PA model does not appear to be well-calibrated with respect to present-day perched water-levels. The source of current perched water may be both precipitation infiltration and service water leakage and thus, the effects of perched water on contaminant flow and transport over long time periods after operations at INTEC cease is uncertain.

The difficulty in calibrating the PA model (DOE Idaho, 2003a) to known perched water levels in the vicinity of TFF is confounded by the lack of knowledge regarding the sources of water that recharge the system. For example, in a recent modeling analysis conducted for the TFF, water inputs and outputs to the system included the following (DOE Idaho, 2006a):

- Infiltration from precipitation
- BLR seepage
- Reinjection in CPP-3 disposal well
- CPP-1 and CPP-2 pumping service water
- CPP-4 and CPP-5 potable water discharges
- Pumping at the Test Reactor Area (TRA)
- Injection at the TRA disposal wells/ponds
- Production in the CFA-1 and CFA-2 water supply wells
- Specified head boundary conditions to represent underflow

Additionally, DOE Idaho did not incorporate recent data into its analysis (i.e., Tc-99 monitoring well data) that show that the conceptual model for contaminant flow and transport may be flawed. The potential ramifications of the hydrogeological conceptual model uncertainty are discussed in Section 4.2.9.1. However, it is significant to note that the RI/BRA modeling for TFF (Rodriguez, 1997), from which the PA model draws heavily, was recently updated (April 2006) following discovery of the Tc-99 plume in the SRPA north of the TFF. The updated three-dimensional modeling is calibrated to recent monitoring and characterization data (calibrated to travel time and concentrations of constituents of concerns), considers transient BLR flow based on hydrograph data from gauging stations for historical seepage rates, uses updated infiltration rates specific for the TFF based on new data and modeling, and considers the geochemical effects of historical releases on future transport of Sr-90 in the vadose zone at the TFF (i.e., effects of cationic species such as hydrogen and sodium ions on surface chemistry and cation exchange capacity of alluvium) (DOE Idaho, 2006e). NRC staff was not cognizant of the updated CERCLA modeling when it met with DOE Idaho on June 1, 2006, to discuss DOE Idaho's groundwater modeling. However, this modeling report (DOE Idaho, 2006e) actually addresses most of the concerns NRC staff expressed to DOE Idaho at the meeting, most notably, calibration of the groundwater model to monitoring data from previous TFF releases.

DOE took credit for significant amount of dilution in its PORFLOW model. Table 12 lists the contribution of natural barriers DOE Idaho took credit for in its PA model (DOE Idaho, 2003a). Based on DOE PA model results (see Figure 21a from DOE Idaho, 2006c), DOE Idaho-predicted contaminant concentrations are reduced four orders of magnitude for non-sorbing, conservative constituents (e.g., Tc-99 and I-129) following release into the vadose zone. For Sr-90, attenuation is more than eight orders of magnitude in the unsaturated and saturated zones (see Figure 21b from DOE Idaho, 2006c) due to dilution, sorption and decay along the flow path to the receptor well location, after the Sr-90 activity is already reduced four to five orders of magnitude due to waste retrieval during “flushing” of the sand pad and due to attenuation and decay during transport through the sand pad and vault floor (peak release occurs 230 years postclosure). Figure 21b (from DOE Idaho, 2006c) shows that for location 4, which represents the shallow sedimentary interbed at the “spillway,” activity concentrations are significantly lower compared to location 2 due to significant attenuation in the interbed (location 2 activity concentrations represent the contaminant concentrations prior to significant transport away from the TFF). Travel between location 4 and location 5 represents attenuation through the basalts, which is significantly lower than the attenuation through the sedimentary interbeds. NRC staff was concerned that DOE took too much credit for dilution from BLR seepage in its PORFLOW model. Table 14 provides NRC’s perspective on the credit DOE could reasonably take for natural system performance given the uncertainties in its model to demonstrate compliance with performance objectives. The actual magnitude of attenuation of Sr-90 in the DOE Idaho PA model was also evaluated against recent monitoring and modeling data (DOE Idaho, 2006d; DOE Idaho, 2006e).

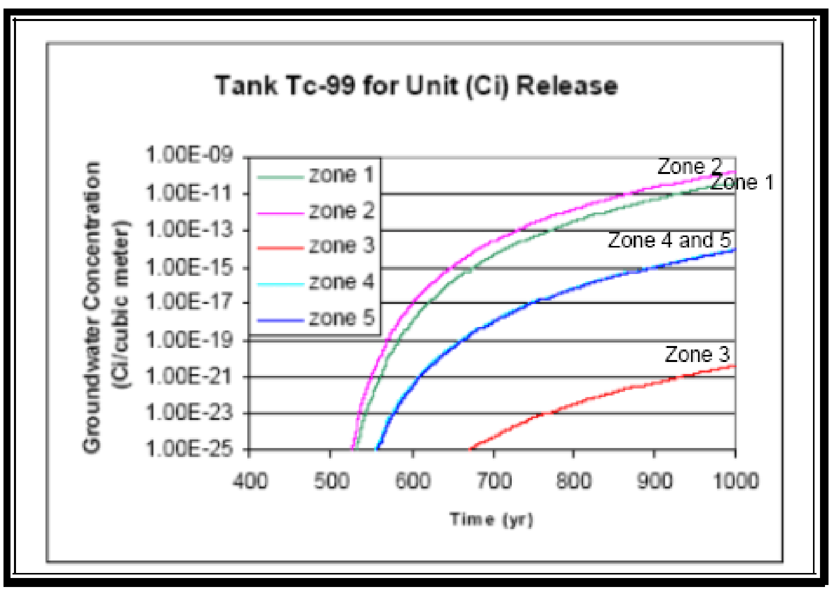

Figure 21a. Tc-99 Concentrations Over Time at Various Monitoring Well Locations

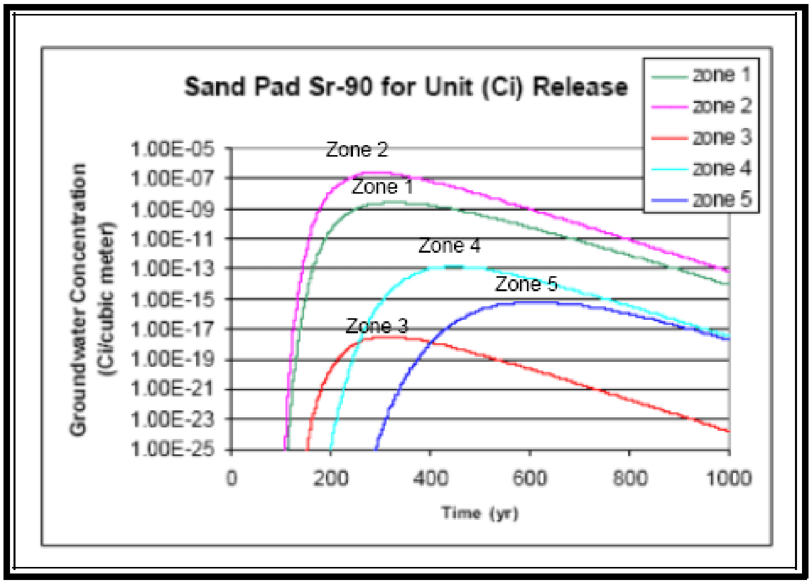

Figure 21b. Sr-90 Concentrations Over Time at Various Monitoring Well Locations

Groundwater monitoring data provides a basis to evaluate DOE model predicted attenuation. The relative concentrations of Sr-90 and Tc-99 in the SBW and in the perched water provide information from which the concentrations of Sr-90 and Tc-99 predicted by the PA model can be assessed. Based on a simplified comparison, the following observations can be made:

- The concentration of Sr-90, Tc-99, and I-129 in the 1972 release from TFF piping into the near-surface alluvium is expected to be approximately 2.3×10^{11} pCi/L, 4.5×10^{7} pCi/L, and 3.6×10^{3} pCi/L, respectively, assuming 70,400 L [18,600 gal] of SBW and the undecayed activities reported in DOE Idaho (Table 5-2, 2006d).

- The current, maximum concentration of Sr-90 detected in monitoring wells screened in the upper perched zone is 2×10^{5} pCi/L (due to attenuation and decay in the alluvium, fractured basalt, and shallow sedimentary interbed located directly underneath the TFF), and maximum concentration of Tc-99 in monitoring well ICPP-MON-A-230 screened in the SRPA is 3×10^{3} pCi/L.

- The concentration of Sr-90, Tc-99, and I-129 in SBW in the year 2012 is expected to be approximately 2.5×10^{10} pCi/L, 9.9×10^{6} pCi/L, and 1.7×10^{4} pCi/L, respectively (Wenzel, 2005).

- The maximum, conservative or compliance case concentrations of Sr-90, Tc-99, and I-129 released from the tanks and sand pad (for Sr-90) at closure are expected to be 7×10^{5} pCi/L (sand pad), 7×10^{4} pCi/L (tank), and 2×10^{3} pCi/L (tank), respectively (assuming the infiltration rate and annual Ci release rates provided in the PA; Tables 3-5 and 4-1, DOE Idaho, 2003a).

- The total inventory of Sr-90, Tc-99, and I-129 released in the 1972 event are 5.9×10^{8} MBq [15,900 Ci], 1.2×10^{5} MBq [3.2 Ci], and 9.3 MBq [2.5×10^{-4} Ci), respectively.

- The total inventory remaining in the tanks (or sand pad for Sr-90) are 2.5×10^7 MBq [678 Ci], 2.2×10^5 [6 Ci], and 222 MBq [6×10^{-3} Ci), respectively.

Therefore, the concentration of Sr-90 that is expected to be released from a single sand pad into the aquifer at the year of peak release is up to five orders of magnitude lower (due to removal of Sr-90 from the sand pad during periodic jet pumping and due to sorption and decay that takes place from the 1962 event to the expected time of peak release at year 2242). The total inventory of Sr-90 in the TFF is also an order of magnitude lower than the 1972 release. The hydraulic and chemical barrier afforded by the grouted vault has a very significant effect on mitigating the release of Sr-90 into the subsurface (4-5 orders of magnitude, see Table 14). Additionally, an attenuation factor of 10 to 100 in the unsaturated and saturated groundwater is reasonable based on calculated flow rates through the aquifer and based on observed dilution of non-sorbing Tc-99 in a saturated zone monitoring well (5×10^7 pCi/L in SBW release compared to 3×10^3 pCi/L in saturated groundwater). The dilution factor based on monitoring data is expected to exaggerate the attenuation capacity of the SRPA, since the monitoring well is not expected to be located in the maximum point of exposure in space and time (the monitoring well data is from a well that was not expected to produce elevated concentrations from historical releases). Thus, the expected Sr-90 concentration in groundwater using a dilution factor of 100 would be 7,000 pCi/L with no credit for sorption and decay. At least another two to three orders of magnitude reduction in activity concentrations due to transport through the vadose zone sedimentary interbeds in the vadose zone directly underneath the TFF should reasonably be expected. A six order of magnitude reduction in Sr-90 concentrations was observed in the perched water underneath the TFF due to attenuation from both the near-surface alluvium above the bottom of the tanks and the upper sedimentary interbed materials underneath the TFF where future releases are expected to be significantly attenuated. Further, the likely bounding inventory for Sr-90 in the sand pad and pessimistic expectations of vault performance provide additional lines of evidence that the TFF can meet performance objectives for Sr-90 (Sr-90 concentrations should be less than 7 pCi/L).

The Tc-99 concentrations released from the grouted tank are expected to be three orders of magnitude lower (due to grout and vault sorption) and the total inventory would be twice as high as the 1972 event. Again, the concentration of Tc-99 is expected to be two orders of magnitude lower due to dilution in the unsaturated and saturated zones, and thus, Tc-99 concentrations may be reasonably expected to be less than 700 pCi/L in saturated groundwater, provided reducing conditions are maintained in the grouted tank or additional dilution or dispersion occurs in the perched water or saturated groundwater.

The I-129 release concentration from the grouted tank is expected to be approximately the same concentration as the 1972 release and the total inventory twice as high. The concentration in saturated groundwater should reasonably be less than 10 pCi/L using a dilution factor of 100 and an attenuation factor of 2 due to sorption in the vadose zone. This concentration would lead to a 4 mrem/yr all-pathways dose from I-129.

Note that the release of radioactive material from the TFF is modeled as a fractional release over time, and the 1972 event represents a point release over a very short time period. Although monitoring data and modeling results for the TFF release may not be directly comparable, this anecdotal evidence suggests that future releases associated with the grouted TFF system are expected to contribute little to the current risks, but may pose a smaller risk over a greater amount of time. While Sr-90 concentrations in the perched zone from the historical release are currently extremely high (2×10^5 pCi/L) and could pose risks even after the end of the institutional control period (assumed to be 2095 in the CERCLA analysis with no remediation), the concrete vault is expected to provide a significant barrier to the release of Sr-90 into the environment, thereby mitigating the potential impact of Sr-90 from future TFF releases. Furthermore, the Sr-90 inventory in the sand pad is likely bounding. Although I-129 in the perched zone underneath the TFF is currently detected in many wells, and above the MCL (1 pCi/L) in one well, the concentration of I-129 released from the tank is expected to be less than the concentration of I-129 released from CPP-31 source in the 1972 event due to tank cleaning activities and due to higher expected attenuation in the grouted tank system at TFF.

4.2.9.4 Evaluation—Flooding Scenario

The impact of a Mackay Dam failure-induced flood on radionuclide transport was analyzed by DOE Idaho. The flood was assumed to occur at tank failure (500-year postclosure). Infiltration was increased by 100 times over the 12.4 cm [4.9 in] per year worst-case scenario infiltration rate. The transport simulations are very sensitive to infiltration rate because it affects not only the transport rate through the vadose zone, but also the release rate from the wasteform (DOE Idaho, 2003a). DOE Idaho modeling results (DOE Idaho, 2003a) indicate that, although the peak concentration arrival time would occur slightly earlier than under non-flooding conditions, the peak concentration estimated under flooding conditions would be less than that estimated for conservative or compliance case (non-flooding) conditions. DOE Idaho explains that the thickness and lateral extent of the perched water bodies beneath the TFF will increase during flooding, thus slowing the movement of radionuclides and allowing more than the normal amount of dilution to occur, given the amount of additional water that would be moving through a flooded system. During the period that follows the peak dose after a flooding event, postpeak concentrations are higher than in the conservative or compliance case scenario, presumably because more radionuclides would be released more quickly from a flooding event. There is some uncertainty associated with the treatment of the flux boundary conditions to represent the wetting front of the ponded water in the flooding scenario and the effect of model construction (e.g., geology, hydrostratigraphic discretization of geology, material properties of model layers, and resultant flow field). However, the conclusion that additional dilution will occur such that the peak dose is reduced is reasonable considering the large amount of water added to the system outside of areas of the TFF that is expected to propagate the perched water and increase dispersion and dilution of the contaminant plume. Local saturated zone water levels are also expected to respond quickly to increased BLR flow, contributing to further dilution and dispersion in the saturated zone.

4.2.9.5 Overall Conclusions

In summary, with respect to hydrology NRC staff has several concerns regarding the DOE Idaho selected conceptual model, including assumptions about the principal direction of vadose zone flow, the BLR boundary condition, and important hydrological features that are thought to be present in the INTEC area but that were not explicitly accounted for in the hydrology model. Staff acknowledge that there is a paucity of site-specific data, including wells and well characterization data, that are directly relevant to the known principle direction of vadose zone flow. At the same time, however, staff were concerned that the implementation of the hydrology model, including the model orientation and methods used to discretize the mapped sedimentary interbeds and basalt flow groups and the BLR and recharge boundary conditions were potentially resulting in an optimistic level of performance by laterally diverting and diluting radionuclides along the flow path to the SRPA. Sensitivity analyses performed by DOE Idaho do not currently address all of these specific concerns. Recommendations related to consideration of recent and future monitoring data and modeling activities are discussed in Appendix A.

Nonetheless, the uncertainties in the hydrologic modeling are offset through consideration of multiple lines of evidence, including sampling data from the tank vaults that suggest the Sr-90 concentrations in the sand pad are likely bounding, consideration of monitoring data from historical TFF releases that provide a basis from which predictive model results can be evaluated for all groundwater HRRs, and consideration of varying levels of performance of the concrete vaults that has the potential to essentially eliminate any concern with short-lived radionuclides releases from the TFF. Therefore, NRC staff has confidence that overall system performance of both the engineered and natural barriers will be sufficient to meet performance objectives.

4.2.10 Dose Methodology

The dose methodology used by DOE Idaho in the performance assessment process was the application of dose conversion factors to an all-pathways exposure scenario. This methodology is widely used in performance assessments and consists of multiplying the radionuclide concentration in air, water, or soil (that a receptor might be exposed to through any of the various pathways) by the dose conversion factor specific to that ingestion or inhalation process and radionuclide.

The calculation process and the dose factors used for the all-pathways exposure performance assessment are described in DOE Idaho (2003b, Section 3). The exposure pathways include drinking water dose from groundwater, all-pathways dose from groundwater (including drinking water dose), air dispersion pathways, and intruder pathways. The primary mechanism for transport of radionuclides from the TFF to a human receptor is expected to be leaching (after degradation of the grouted tanks and vaults) to the groundwater and human consumption and use of well water for domestic purposes. The exposure pathways considered in the performance assessment involving contaminated well water include direct ingestion, ingestion of milk and meat from dairy and beef cattle consuming contaminated well water, and ingestion of plants and animals grown and raised in areas irrigated with contaminated well water.

In the intruder analysis performed by DOE Idaho for the PA, radionuclide dose conversion factors from the Federal Guidance Reports developed by the EPA (1993, 1988) were used (DOE Idaho, Section 5, 2003a). Ingestion and inhalation dose conversion factors were taken from Federal Guidance Report 11 (EPA, 1988), and external dose conversion factors were taken from Federal Guidance Report 12 (EPA, 1993). Federal Guidance Report 11 provides 50-year committed effective dose equivalents per unit of activity based on the exposure pathway (inhalation or ingestion) and the specific radionuclide. The intruder analysis dose conversion factors were included in the performance assessment document (DOE Idaho, Table 5-1, 2003a).

4.2.11 NRC Evaluation—Dose Methodology

The dose methodology implementation of the performance assessment is well supported and suited for the purpose. Numerous NRC guidance documents recommend the approach and use of the specific dose conversion factors used in the performance assessment process. These include NUREG–1573 (NRC, 2000), which provides guidance on the use of pathway dose conversion factors for calculating doses via the potential exposure pathways, and NUREG–1757 (NRC, 2003, Volume 2, Appendix I), which provides guidance on the use of specific dose conversion factors such as those developed by EPA and published in Federal Guidance Reports 11 and 12 (EPA, 1993, 1988). These guidance documents and their recommended methodologies were used by DOE Idaho in the development of the performance assessment and provide confidence that the dose methodology implementation is appropriate for comparison to the performance objectives.

4.2.12 Protection of the Public

The public is represented by an adult member of a farming community that lives in a residence downstream of the existing TFF (the resident-farmer scenario). During the operational and institutional control periods, it is assumed that the individual resides at the INL site boundary. After active institutional controls cease at 100 years, the member of the public resides at the INTEC facility. An offsite member of the public is assumed to use water from a well for domestic purposes after the institutional control period. The well is assumed to be located where the maximum concentration of radionuclides in the ground water is predicted to occur. NRC guidance (NRC, 2006b) specifies a public receptor location just outside a buffer zone extending approximately 100 m [330 ft] from the disposal area. In the case of the TFF, the tanks are expected to be regarded as disposal units, and an appropriate buffer zone is expected to extend 100 m [330 ft] from the line circumscribing the tanks. The guidance provides that in some instances, such as with a complex hydrogeologic system or where there are multiple sources, the point of maximum exposure may be at a larger distance than the 100 m [330 ft] distance from the disposal unit. For the INTEC TFF, maximum contaminant concentrations are, in fact, predicted 600 m [2,000 ft] downgradient from the facility in the PA, primarily as a result of BLR seepage contributions to pressure gradients in the vadose zone (DOE Idaho, 2006c). Contaminant concentrations are not diluted as a result of extraction of contaminated water with the well. However, contaminant concentrations are averaged over a 10-m [33-ft] well-screen length.

The evaluated exposure pathways include the ingestion of contaminated water, ingestion of contaminated food, and inhalation of contaminated airborne particulates. Release into the air pathway of volatile radionuclides was considered separately. The analysis of the exposure pathways indicates that the groundwater pathway was the most significant in terms of radionuclide transport to the receptors. The methodology used to calculate the all-pathways dose is based on the methodology present in reports by NRC (1977), Peterson (1983), and Maheras, et al. (1997). Parameters used in the dose model were primarily derived from values for the Yucca Mountain Project because the climate and geography are somewhat similar (LaPlante, 1997). To account for uncertainty in the dose assessment modeling, most biosphere parameters are stochastic. DOE Idaho used the 95 percent confidence level for comparison to the performance objectives. Dose conversion factors used were taken from Federal Guidance Reports 11 and 12 (EPA 1988, 1993).

The all-pathways total effective dose equivalent (TEDE) to a member of the public was predicted to be 0.014 mSv/yr [1.4 mrem/yr] at approximately 890 years, which does not exceed the Part 61 limit of 0.25 mSv/yr [25 mrem/yr] to the whole body (DOE Idaho, 2003a).[16] Over 99 percent of the dose was from I-129 and Tc-99, with much smaller contributions from Sr-90. DOE Idaho applied a compliance period of 1,000 years as per the requirements of DOE Order 435.1 and its associated manual and guidance (DOE, 1999). An evaluation was also performed for time periods to 1 million years to assess longer-term impacts, and the peak all-pathways annual dose from the more slowly transported radionuclides was less than the early annual

[16]The dose methodology used in 10 CFR Part 61, Subpart C (based on International Commission on Radiological Protection Publication 2 (ICRP 2)), is different from that used in the newer ICRP 26. However, use of the newer ICRP 26 methodology is consistent with NRC policy (NRC, 2002a).

dose {e.g., 0.014 mSv/yr [1.4 mrem/yr]} from I-129. The modeling results from the PA (DOE Idaho, 2003a) are presented in Table 8.

The results from the PA were scaled to the ratio of the inventory developed in the waste determination (DOE Idaho, 2005) to the inventory developed for the PA (DOE Idaho, 2003a). The adjusted peak dose based on the revised inventory is 0.005 mSv/yr [0.5 mrem/yr] to a member of the public from Tc-99. See Table 9 for a summary of the results.

4.2.13 NRC Evaluation—Protection of the Public

DOE Idaho used an all-pathways dose assessment to show conformance with the performance objectives established for the public. The peak TEDE to a member of the public of 0.014 mSv/yr [1.4 mrem/yr] based on the PA modeling (DOE Idaho, 2003a) is well within the performance objective of 0.25 mSv/yr [25 mrem/yr] in 10 CFR 61.41 ("Protection of general population from releases of radioactivity"). The predicted dose based on the revised inventory presented in the waste determination (DOE Idaho, 2005) is expected to be less than 0.01 mSv/yr [1 mrem/yr].

The contribution of key (or highly radioactive) radionuclides to peak annual dose and the timing of the peak annual dose can be significantly influenced by uncertainties. Because the PA results were deterministic (with the exception of the dose model), DOE Idaho provided a series of sensitivity analyses to evaluate the impact of key uncertainties. The key uncertainties were residual radionuclide inventory, infiltration rate, transport parameters, and grout distribution coefficients. Table 10a contains a summary of the sensitivity analyses results. The shaded row is the result DOE Idaho used to compare to the performance objectives. Additional sensitivities were evaluated but were not included in the matrix of all-pathways sensitivity analyses.

The matrix on sensitivity results provided in Table 10a represents 24 distinct deterministic analyses. The inventory was assigned four uncertainty scenarios (worst, conservative, realistic, and best). The other three main areas evaluated (Grout K_d, Transport, Infiltration) were each assigned three uncertainty scenarios. Ideally, this would produce 4 × 3 × 3 × 3 or 108 analyses. However, the source term (grout K_d) and transport uncertainties were not varied independently; therefore, the number of analyses becomes 4 × 3 × 3 or 36 (the lowest infiltration rate is not shown, only 24 are shown). As an example, the eighth row in Table 10a represents worst grout K_d, worst transport, an infiltration rate of 4.1 cm/yr [1.6 in/yr] and the best inventory. The total dose column shows that under different conditions, different radionuclides will dominate the total annual dose, and total dose is not a summation of radionuclide-specific doses because of variability in the arrival times.

Almost all risk associated with Sr-90 is from the contaminated sand pads under two of the tanks. The arrival time for Sr-90 to the dose receptor ranged from 294 years for the worst-case results to 1,310 years for the best-case results. As a result of radioactive decay, every 100 years of delay in arrival, either from the engineered system or the geologic system, results in a reduction in activity of approximately a factor of 12 in the Sr-90 risk. Only for pessimistic parameter selection for all main uncertainties would the system not meet the performance objectives {0.85 mSv/yr [85 mrem/yr] from Sr-90}. However, as discussed in Sections 4.2.4 and 4.2.5, the highest infiltration rate used in DOE Idaho's sensitivity analysis, 12.4 cm/yr [4.9 in/yr], is less than the assumed 18 cm/yr [7.1 in/yr] in the latest modeling performed for the TFF (DOE Idaho, 2006e). Finally, as noted in Sections 4.2.6 and 4.2.7, dilution caused by the BLR seepage boundary is likely overestimated, and this uncertainty is not addressed in the sensitivity analysis. However, due to the likely conservatism present in the estimated inventory for the sand pad and concrete degradation modeling, it is expected that Sr-90 will actually decay substantially, and any significant releases from the vault will be attenuated in the INTEC subsurface as supported by monitoring data discussed in Section 4.2.9 above (DOE Idaho, 2006c).

The arrival times for I-129 ranged from 538 years to 5,670 years in DOE Idaho's sensitivity analyses (DOE Idaho, 2003a). Similar to Sr-90, the model results suggest that only under a very pessimistic scenario would the system not meet the performance objectives. However, again the infiltration rates may not be overly pessimistic unless infiltration controls are put into place and the I-129 concentrations are comparable to the I-129 releases from the 1972 event. The concentrations of I-129 in perched groundwater show elevated levels with only one well exceeding the MCL of 1 pCi/L.

The timing of release for Tc-99 is controlled by the K_d used for the grout and concrete. Tc-99 is a non-sorbing, conservative constituent in the vadose zone; therefore, the timing of release is immaterial to the concentrations of Tc-99 that eventually reach the saturated zone, although the magnitude of the release is affected by the Tc-99 grout K_d. Therefore, the main concerns regarding the Tc-99 predictions are related to the assumption of reducing conditions (grout K_d) and potential dilution of Tc-99 in the vadose and saturated zones through dispersion. A comparison of the activity concentrations of Tc-99 expected to be released in the subsurface from the grouted tanks versus the activity concentrations estimated from the 1972 release (see Section 4.2.9.3) provides additional support that Tc-99 can meet performance objectives. Note, however, that the predicted Tc-99 releases are mitigated by the reducing grout that provides a very significant barrier to the release of Tc-99.

Cleaning results performed to date have achieved residual levels of waste significantly lower than expected. The actual risk to the public from tank residuals (primarily from Tc-99 and I-129) is likely to be significantly lower (see updated doses provided in Table 9). Additional sensitivity analyses were conducted for the sand pad inventory that shows the inventory can be a factor of 8 higher. For Sr-90, tank cleaning is immaterial to the dose calculations, because Sr-90 risk is from the sand pad which was not cleaned. As discussed in Section 3.1, the Sr-90 inventory is expected to be bounding considering indirect sampling data from the vault sumps that receive drainage from the sand pad.

Staff concludes that 10 CFR 61.41 performance objectives can be met, including the provision that reasonable effort should be made to maintain releases of radioactivity to the general environment as low as is reasonably achievable (ALARA). The ALARA provision is not part of the PA calculation because the PA generates results to compare to performance objectives. Through demonstration of Criterion Two (the waste has had HRRs removed to the maximum extent practical) and efforts to stabilize the wastes within the engineered grout pour and encapsulation pour, DOE Idaho satisfied the intent of the provision to maintain releases of radioactivity to the environment ALARA. It should be noted that DOE Idaho evaluated the commonly-used resident farmer scenario to assess the public exposures.

4.2.14 Protection of Intruders

A receptor engaging in activities on the disposal site, rather than outside the buffer zone (see Section 4.2.12), is regarded as the inadvertent intruder for demonstrating compliance with 10 CFR 61.42 (NRC, 2006b). DOE Idaho analyzed four intruder scenarios. Many of the standard scenarios were not considered to be applicable to the tanks because depth to the waste in the tanks is 10 m (33 ft) or more. The only applicable scenarios were an intruder-drilling scenario for residual waste in the tanks, an intruder-construction scenario for piping, and an intruder-discovery scenario for piping. The intruder-discovery scenario consequences were bounded by the intruder-construction scenario because of exposure time differences, and therefore, it was not necessary to retain the intruder-discovery scenario for further analysis. DOE Idaho evaluated (DOE Idaho, 2003a) acute and chronic radiological impacts associated with both scenarios (intruder drilling and intruder construction). Approximately 1,000 m [3,300 ft] of process piping will be within 3 m [10 ft] of the land surface. The analyses used a 100-year period for active institutional controls. During this time, fences and armed patrols will prevent inadvertent intrusion.

It is difficult to predict future actions of humans over hundreds to thousands of years. The intruder analyses assume that humans will disrupt the waste at 100 years, with no consideration of the likelihood of occurrence. The risks from human intrusion are very sensitive to the time of intrusion because the short-lived fission products (e.g., Cs-137) are the main contributors to the intruder doses. Uncertainty exists in the state of concrete systems over time. However, DOE Idaho believes that credit could be taken for reinforced concrete vaults and stainless steel tanks, further reducing the doses for the intruder. For the intruder analyses, every attempt was made to consider the site-specific environment and habits of the people currently in the region.

For the intruder-drilling scenario, an irrigation well or domestic drinking water well is drilled directly through the waste. The acute intruder is exposed to drill cuttings spread on the land surface. Exposure time is set at 160 hours compared to the typical value of 6 hours to account for the difficulty of developing an irrigation well at INL due to the presence of basalts in the subsurface. The assumed diameter of an irrigation well is 0.56 m (1.84 ft) and the diameter of a residential drinking water well is 0.15 m [0.5 ft]. Well diameters are derived from site-specific observations. For the acute intruder-drilling scenario, the maximum dose occurs in the first year after the institutional control period ends and is 2.32 mSv [232 mrem] using the conservative inventory and 1.52 mSv [152 mrem] using the waste determination inventory. The major radionuclide contributors are Cs-137 from the external dose pathway and Am-241 and Pu-238 from the inhalation pathway. Chronic exposure is considered as an extension of the acute drilling scenario. It is assumed that the intruder occupies the site after drilling a water well and grows crops on a mixture of clean soil and contaminated drill cuttings. Analyzed exposure pathways include inhalation of resuspended drill cuttings and ingestion of beef, milk, and vegetables contaminated via drill cuttings, but did not include the groundwater pathway, as this is evaluated separately. The maximum dose for the chronic intruder post-drilling scenario occurs in the first year after the institutional control period ends and is 0.911 mSv/yr [91.1 mrem/yr] with Sr-90 and Cs-137 as the main contributors to dose. The expected dose is 0.25 mSv/yr [25 mrem/yr] using the updated inventory in the waste determination (DOE Idaho, 2005).

The intruder-construction scenario involves an inadvertent intruder who excavates or constructs a building on the disposal site. In this scenario, the intruder is assumed to dig a 20- × 10-m [70- × 30-ft] basement to a depth of approximately 3 m [10 ft]. It is assumed that the intruder does not recognize the hazardous nature of the material that is excavated. Acute exposures occur from inhalation of resuspended contaminated soil, ingestion of contaminated soil, and external radiation from contaminated soil. The maximum dose for the acute intruder-construction scenario occurs in the first year after the institutional control period ends and is 0.008 mSv [0.8 mrem] or 0.0023 mSv/yr [0.23 mrem/yr] with the updated inventory in the waste determination. Chronic exposures were also considered by evaluating an intruder who lives in a building constructed as part of the intruder-construction scenario, engages in agricultural activities on the contaminated site, and is exposed to contamination through external irradiation, inhalation of excavated contaminated soil, inhalation of gaseous radionuclides, ingestion of soil, and ingestion of contaminated beef, milk, and vegetables that were produced at the site. The maximum dose for the chronic intruder post-construction scenario occurs in the first year after institutional control period ends and is 0.261 mSv/yr [26.1 mrem/yr] or 0.032 mSv/yr [3.2 mrem/yr] with the updated inventory provided in the waste demonstration, with Cs-137 as the main contributor to dose.

A numerical performance objective is not provided in 10 CFR 61.42; however, a dose limit of 5 mSv [500 mrem] per year was described in the Draft Environmental Impact Statement for Part 61 for development of waste classification requirements and is applied here for intruder scenarios (NRC, 1981). All intruder scenario doses are less than 5 mSv [500 mrem] per year (all-pathways TEDE).

4.2.15 NRC Evaluation—Protection of Intruders

The NRC staff was not convinced that DOE Idaho used appropriate assumptions in developing the well intruder scenario. The area that the waste was assumed to be spread over in the acute driller scenario [0.5 acres (2,200 m^2)] led to a very thin layer of contamination of 1.4 cm [0.54 in]. Because it is not reasonable to assume that the drill cuttings would be spread over this area during well drilling and development, NRC staff asked DOE Idaho to evaluate the sensitivity of the results to various parameter values used in the intruder calculation. Furthermore, the uncertainty in the sand pad inventory was not taken into consideration when these calculations were performed. In response to NRC's RAI 16 (NRC, 2006a), an intruder sensitivity analysis was conducted for both the acute and chronic intruder-drilling scenarios (DOE Idaho, 2006a). The acute intruder-drilling scenario assumed that the cuttings from a 0.6-m [22-in] irrigation well were placed in a 3- by 3-m [10- by 10-ft] pit during drilling operations. The intruder was assumed to stand next to the pit for the entire drilling duration of 160 hours. The waste residuals from the tank and sand pads would initially be exposed at the surface of the pit, and then increasing thicknesses of clean material would be placed on top of the contaminated material. DOE Idaho used MicroShield to evaluate the external dose contribution for various thicknesses of clean cover material. A constant drilling rate was assumed over the total drilling time of 160 hours. Because the depth to groundwater is 122 m [406 ft], it would take approximately 17 hours to reach the tank heel. DOE Idaho modeled increments of clean cover {assumed to be 10 cm [4 in][17]} which would take about 1.3 hours per increment (DOE Idaho, 2006a). The results show that the external dose from Cs-137 could be a factor of two higher with this alternative model and a factor of three higher considering the uncertainty in the sand pad inventory. Considering both of these factors, the intruder dose could be higher than the performance objective of 5 mSv/yr [500 mrem/yr]. However, this would only occur under very pessimistic conditions.

A sensitivity analysis was also conducted for the chronic intruder-drilling scenario. Three parameters were varied: well diameter {25, 20, and 15 cm [10, 8, and 6 in]}, contaminant spreading area {1,100, 1,650, and 2,200 m^2 [0.3, 0.4, and 0.5 acres]}, and the tilling depth {30, 46, and 61 cm [12, 18, and 24 in]}. The results of the sensitivity analysis for the chronic intruder-drilling scenario show that the dose of 0.21 mSv/yr [21 mrem/yr] reported in the PA (DOE Idaho, 2003a) could be a factor of 3 higher with a larger diameter well {25 cm [10 in] versus 15 cm [6 in]}, a factor of 2 higher with a smaller area of contamination {1100 m^2 [0.3 acres]}, a factor of two higher considering the highest inventory in the sand pad, and a factor

[17]There appears to be an error in the RAI 16 response (DOE Idaho, 2006a). DOE Idaho states that it would take approximately 0.5 hr/cm [1.3 hr/in] of cover. However, using the well radius of 0.6 m [22 in] and the pit area of 9.3 m^2 [100 ft^2], it should take approximately 1.3 hr/m [0.4 hr/ft] of borehole and 0.12 hr/cm [0.3 hr/in] of cover. Thus, it should take about 1.3 hours for 10 cm [4 in] of cover or about 1 m of borehole.

of two higher considering a factor of two reduction in the tilling depth. All of these factors considered would lead to a maximum dose just over the 500 mrem/yr performance objective. Again, only under very pessimistic circumstances could the dose to an inadvertent intruder approach the performance objective of 500 mrem/yr for the chronic well drilling scenario.

With the additional sensitivity analyses, DOE Idaho developed reasonable intruder scenarios to evaluate protection of inadvertent intruders and demonstrated that performance objectives found in 10 CFR 61.42 ("Protection of individuals from inadvertent intrusion") could be achieved. Acute and chronic exposures associated with intruder-drilling scenarios resulted in significantly larger doses than the intruder-construction scenarios. All intruder doses are calculated to be less than 5 mSv/yr [500 mrem/yr], except under very pessimistic conditions. Once the remaining uncleaned tanks are cleaned, they should be sampled to establish a final post cleaning inventory and this inventory should be compared to the bounding inventory used to calculate the intruder scenario doses above.

4.2.16 Protection of Individuals During Operations

The performance objective in 10 CFR 61.43 cross-references the standards for radiation protection in Part 20. DOE's approach to demonstrating protection of individuals during operations (10 CFR 61.43) is to cross-walk the relevant DOE regulation or limit with that provided in 10 CFR Part 20 and demonstrate that the DOE regulation provides an equivalent level of protection. The cross-referenced "standards for radiation protection" in 10 CFR Part 20 that are considered in detail in Section 7.12 of DOE Idaho (2005) are the dose limits for the public and the workers during disposal operations set forth in 10 CFR 20.1101(d), 10 CFR 20.1201(a)(1)(i), 10 CFR 20.1201(a)(1)(ii), 10 CFR 20.1201(a)(2)(i), 10 CFR 20.1201(a)(2)(ii), 10 CFR 20.1201(e), 10 CFR 20.1208(a), 10 CFR 20.1301(a)(1), 10 CFR 20.1301(a)(2), and 10 CFR 20.1301(b). These dose limits correspond to the dose limits in 10 CFR Part 835 and relevant DOE Orders which establish DOE regulatory and contractual requirements for DOE facilities and activities.

A number of measures will ensure that exposure of individuals during operations are maintained ALARA. These include (i) a documented Radiation Protection Program (RPP) that includes formal plans and measures for applying the ALARA process to occupational exposure; (ii) a Documented Safety Analysis (DSA) that identifies, classifies, and evaluates hazards associated with the TFF closure project; (iii) design; (iv) regulatory and contractual enforcement mechanisms; and (v) engineered and access controls, training, and dosimetry (DOE Idaho, 2005). The discussion that follows was provided in Section 7.12 of DOE Idaho (2005).

Radiation exposures recorded for all TFF work performed in association with tank cleaning from January 1, 2002, through June 15, 2005, provide a total of 49.31 person-mSv [4,931 person-mrem]. Maintenance-related process line and valve work for this period accounted for 25.68 person-mSv [2,568 person-mrem]. The remaining 23.63 person-mSv [2,363 person-mrem] was attributed to cleaning 7 tanks, yielding an average of 3.38 person-mSv [338 person-mrem] per tank. Based on this average of the 7 cleaned tanks versus the actual exposure of cleaning the most contaminated tank (WM–182) of 6.11 person-mSv [611 person-mrem], a reasonable dose projection for cleaning the 4 remaining tanks has been provided.

The effectiveness of the radiation protection programs at the TFF has been demonstrated by past occupational exposure results. Worker dose for tank cleaning is minimal because cleaning is accomplished remotely. Worker exposures are limited to equipment installation and operation and maintenance activities on contaminated equipment. Based on the tank cleaning dose history discussed above, worker exposure is estimated to total approximately 6.5 mSv [650 mrem] for 23 workers for an average exposure of 0.3 mSv [30 mrem] per person. The maximum radiation exposure for any single worker is estimated to be 1.2 mSv [120 mrem] for cleaning a single TFF tank of the 4 that remained to be cleaned. This estimate comes directly from the maximum exposure any worker has received from TFF closure activities to date of 1.17 mSv [117 mrem]. These dose estimates compare to a 10 CFR 20.1201(a) and 10 CFR Part 835 limit of 50.0 mSv per year [5,000 mrem per year].

The air pathway is the predominate pathway for doses to the public from TFF cleaning operations. These doses are estimated to be 0.0051 mSv per year [0.51 mrem per year] (DOE Idaho, 2003a), well below the 0.1 mSv [10 mrem] annual limit.

4.2.17 NRC Evaluation—Protection of Individuals During Operations

DOE has provided adequate information that individuals will be protected during operations. DOE provided a cross-walk of the relevant DOE regulations to those provided in 10 CFR Part 20, which is incorporated as part of 10 CFR 61.43 performance objective. NRC staff agrees that an equivalent level of protection is provided by the relevant DOE regulations or limits to the requirements found in 10 CFR 20.1101(d), 10 CFR 20.1201(a)(1)(i), 10 CFR 20.1201(a)(1)(ii), 10 CFR 20.1201(a)(2)(i), 10 CFR 20.1201(a)(2)(ii), 10 CFR 20.1201(e), 10 CFR 20.1208(a), 10 CFR 20.1301(a)(1), 10 CFR 20.1301(a)(2), and 10 CFR 20.1301(b). In addition, a number of measures are applied to ensure that exposure to individuals is ALARA including (i) a documented RPP, (ii) a DSA, (iii) design, (iv) regulatory and contractual enforcement mechanisms, and (v) engineered and access controls, training, and dosimetry.

The public will be located a significant distance (several kilometers) from the facilities during operations, and active security is maintained to prevent inadvertent access to the site. The NRC staff agrees with DOE Idaho that the risk to the public during operations should be minimal, and the relevant limits can be achieved.

4.2.18 Site Stability

The four 100-m^3 [30,000-gal] tanks are underground on gravel-filled reinforced concrete pads outlined with a curb. The tanks have a diameter of approximately 3.5 m [11.5 ft] and are buried by compacted gravel; they are located at a depth of approximately 8.8 m [29 ft] below the surface. These tanks are covered with approximately 3 m [10 ft] of soil. The eleven 1,000-m^3 [300,000-gal] storage tanks are contained in underground concrete vaults. The vault floors are approximately 14 m [45 ft] below ground. The tank wall heights are 6.4 m [21 ft], with the exception of two tanks that have 7.0-m [23-ft]-high walls. DOE Idaho plans to fill the piping, vaults, and tanks with grout. The current depth to residual waste in the tanks is greater than 10 m [33 ft], and the depth to residual waste in the process piping is greater than 3 m [10 ft] for 70 percent of the process piping. Approximately 30 percent of the process piping is within 3 m [10 ft] of the land surface. The process piping will be filled with grout upon closure of the facility to ensure structural stability.

The structural stability of the TFF relies on grouting the space between the tank and the concrete vault and the space inside the tank. Grout will create a solid monolith with little void space and eliminate differential settlement, which is commonly observed in low-level waste disposal vaults. Although degradation of the grouted tanks and vaults may be expected over long periods of time, significant structural collapse is not anticipated, so structural failure (collapse) is not considered in the PA.

The TFF is located in an area of relatively low seismic activity. The most significant sources of seismic hazard in the INTEC vicinity are (i) the Lemhi fault where a magnitude 7.15 earthquake occurred at the southern end, (ii) the Lost River fault where a magnitude 7.25 earthquake occurred at the southern end, and (iii) either the Arco or the Lava Ridge–Hell's Half Acre volcanic rift zone and the axial volcanic zone where a magnitude 5.5 earthquake associated with dike injection occurred. Background seismic activity has also occurred in the Eastern Snake River Plain and the northern Basin and Range Province (Woodward-Clyde Federal Services, 1996a). Contributions to the overall seismic hazard from other sources (including the postulated ESRPlain boundary fault, northern Basin-and-Range Province, Yellowstone Plateau, and Idaho Batholith) are significantly lower because they are farther from the site and generally have smaller maximum magnitudes. The design basis ground motion for the Idaho Spent Fuel Facility at INTEC was based on the horizontal mean of 5-percent damped uniform hazard spectra for rock at 2,000- and 10,000-year return periods, as developed by URS Greiner, et al. (1999). A seismic evaluation of the Idaho Spent Fuel Facility showed that the surficial alluvium is not susceptible to liquefaction because the deposits are coarse and located well above the water table (NRC, 2004).

Holocene geologic and archeologic studies suggest that fluvial and eolian deposition and tectonic subsidence at INL have been in approximate net balance for at least 10,000 years and that the INL area will likely continue its pattern of regional subsidence and net accumulation of sedimentary and volcanic materials, although sedimentation patterns will change in response to future climate fluctuations (DOE Idaho, 2003a). A reversal of the regional pattern of Eastern Snake River Plain subsidence, sedimentation, and volcanism into an erosional rather than a depositional regime would require major changes from the Holocene tectonic or climatic configuration. Erosion of soil covering the TFF that could occur as a consequence of faulting and uplift of the southern Lost River Fault if the fault were to encroach southward to a position several kilometers west of INTEC would necessitate a major change in the tectonic configuration of the Eastern Snake River Plain. This scenario is therefore considered improbable within the next 10,000 years (DOE Idaho, 2003a). Future climate fluctuations (to either colder and wetter or warmer and drier conditions) are not expected to erode the INTEC land surface.

Based upon DOE Idaho analyses that assume fluvial sediments do not accumulate and infill the local depressions of the Flood Diversion Facility, the impact of the probable maximum flood on the TFF is expected to be minimal (DOE Idaho, 2003a) because the surface elevation of the TFF is near the highest elevation to which floodwaters are anticipated to rise. The distance of the tanks from the BLR (460 m or 1,500 ft) and lack of significant grades are expected to limit fluvial erosion. Because the facility is near the edge of the estimated floodwaters, surface water flow velocities are expected to have minor erosional effects. One to 2 m [3.3 to 6.6 ft] of water could cover the facility for a short period of time, resulting in a wetting front to infiltrate the subsurface at TFF and potentially transport large quantities of water that could mobilize constituents in a degraded disposal system.

4.2.19 NRC Evaluation—Site Stability

The TFF will be closed to achieve long-term stability of the disposal site and to eliminate the need for ongoing maintenance to the extent practicable. DOE Idaho plans to fill the tanks and vaults with 10 m (33 ft) or more of grout to indicate that safety requirements comparable to

10 CFR 61.44 ("Stability of the Disposal Site after Closure") can be met. The grouted tanks and vaults will likely contain minimal void space, eliminating differential settlement and the associated negative effects on waste isolation. Future actions to close the TFF will include infiltration controls and some type of engineered barrier (DOE Idaho, 2003a, 2006e). DOE Idaho has not developed a cover design at this time and therefore takes no credit in the PA (DOE Idaho, 2003a) for a robust engineered cover that may limit infiltration of water to the waste, although it takes credit for an earthen cover comparable to the cover found at the CFA in its compliance case. An engineered barrier could enhance site stability by minimizing aeolian (wind-driven) erosion.

Aeolian erosion can be significant in arid environments, but it is not expected to be significant for this incidental waste determination primarily due to topography and the depth of the waste. The topographic relief of the TFF and surrounding portions of INTEC is minor, and thick soil (i.e., 3 m or 10 ft) covers the tanks. The TFF is located in the flood plain of the BLR. DOE Idaho use of the PMF from a PMP-induced failure of Mackay Dam is considered by the NRC staff to provide a reasonable basis for the PA (see Section 4.2.9) because the arid environment and lack of significant grades are expected to limit fluvial erosion, although the PMF may temporarily inundate much of the surface of the TFF to shallow depths [0 to 1 m (0 to 3 ft)]. Erosion is thus unlikely to influence the stability of the disposal site.

DOE Idaho has presented sufficient information to conclude that the design can adequately provide long-term stability of the wasteform for erosion control purposes. The intruder construction scenario assumes exposure starting immediately at the loss of institutional controls (i.e., 100 years), and erosion processes would not be expected to expose the process piping in such a short period of time (if at all) based on site conditions. Therefore, the intruder-construction scenario with chronic exposure of less than a 0.01-mSv/yr [1-mrem/yr] annual dose provides a reasonable bound to any potential erosion concerns associated with the process piping at INTEC.

4.3 NRC Review and Conclusions (Criterion Three)

The following assumptions were used in assessing conformance with Criterion Three:

- Active institutional controls will be maintained for 100 years.
- The model limitations or uncertainties NRC staff identified in DOE Idaho's hydrogeological conceptual model (HCM) and hydrogeologic model construction and implementation will not significantly alter the conclusions in this TER.
- Inventory estimates for the large tanks that have not been cleaned (WM–187 through WM–190) are not significantly underpredicted (i.e., similar or better waste retrieval will be achieved than is currently assumed by DOE Idaho).

NRC staff's conclusions with respect to Criterion Three are the following:

There is reasonable assurance that DOE Idaho can meet Criterion Three of the NDAA because:

- Based on information provided by DOE Idaho, NRC staff expects the maximum public dose from all pathways to be below the 0.25-mSv/yr [25-mrem/yr] dose limit. Reasonable effort will be made by DOE Idaho to maintain releases of radioactivity in effluents to the general environment as low as reasonable achievable. Therefore NRC staff concludes that there is reasonable assurance that DOE Idaho can meet 10 CFR 61.41 requirements.

- Based on analysis provided by DOE Idaho, NRC staff concludes there is reasonable assurance that DOE Idaho can meet 10 CFR 61.42 requirements for protection of individuals from inadvertent intrusion.

- Workers are protected by DOE regulations that are comparable to 10 CFR Part 20. DOE Idaho controls are also in place to protect members of the public during operations. Therefore, the NRC staff concludes that there is reasonable assurance that DOE Idaho can meet 10 CFR 61.43 requirements for protection of individuals during operations.

- DOE Idaho plans to fill the tanks, vaults, and ancillary equipment with grout which will provide structural stability and limit waste dispersal. Therefore, NRC staff concludes that there is reasonable assurance that DOE Idaho can meet 10 CFR 61.44 stability requirements.

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

The NRC staff has identified the following list of key monitoring areas related to the features of engineered and natural system performance of the TFF disposal facility that are important to demonstrating compliance with the performance objectives in 10 CFR Part 61, Subpart C. The NRC will coordinate with the State of Idaho to develop a monitoring plan/approach addressing the following:

- DOE should sample tanks WM-187 through WM-190 after cleaning as stated in Section 2.3 of the Draft Section 3116 Determination Idaho Nuclear Technology and Engineering Center Tank Farm Facility (DOE Idaho, 2005). Sampling data and analysis of tanks WM–187 through WM–190 after cleaning should be reviewed to ensure that the inventory for these tanks is not significantly underestimated (i.e., similar or better waste retrieval will be achieved).

- The final grout formulation used to stabilize the TFF waste should be consistent with design specifications or significant deviations should be evaluated to ensure that they will not negatively impact the expected performance of the grout. The reducing capacity of the tank grout is important to mitigating the release of Tc-99. Short-term performance of as-emplaced grout should be similar to or better than that assumed in the PA release modeling or significant deviations should be evaluated to determine their significance with respect to the conclusions in the PA and this TER. The short-term performance of the grouted vault is especially important to mitigate the release of short-lived radionuclides such as Sr-90 from the contaminated sand pads that could potentially dominate the predicted doses from the TFF within the first few hundred years (DOE Idaho, 2003a).

- Relevant recent and future monitoring data and modeling activities should continue to be evaluated to ensure that hydrological uncertainties that may significantly alter the conclusions in the PA and this TER are addressed. If significant new information is found, this information should be evaluated against the PA and TER conclusions.

- Closure and postclosure operations (until the end of active institutional controls, 100 years) will be monitored to ensure that the 10 CFR Part 61.43 performance objective (protection of individuals during operations) can be met. As part of this assessment, radiation records, environmental monitoring, and exposure assessment calculations may be reviewed.

- INTEC infiltration controls and the construction and maintenance of an engineered cap over the TFF under the CERCLA program should be monitored to ensure that PA assumptions related to infiltration and contaminant release are bounding.

Performance indicators may be developed by NRC in conjunction with the State of Idaho to evaluate system performance. As additional information is obtained, NRC will coordinate with the state to adjust the scope, as appropriate, of monitoring activities related to TFF waste disposal at INL.

5 OVERALL CONCLUSIONS

As discussed in detail in previous sections, NRC staff has conducted a technical analysis of DOE Idaho's waste determination for TFF waste at the INL. The NRC staff concludes that DOE Idaho has adequately demonstrated that NDAA criteria in Section 3116 (a)(1), (a)(2), and (a)(3)(A)(i) or (a)(3)(B)(i) can be met for residual waste disposed of at TFF. This conclusion is based on information presented in DOE Idaho's draft Section 3116 Waste Determination dated September 7, 2005; DOE's responses to NRC's RAI; supporting references; and information provided during meetings between NRC and DOE. The NDAA requires NRC, in coordination with the State of Idaho, to monitor disposal actions taken by DOE to assess compliance with the performance objectives in 10 CFR 61, Subpart C. NRC will continue to coordinate with the Idaho DEQ to develop a program by which NRC and the state will monitor DOE's disposal actions.

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

APPENDIX A
RECOMMENDATIONS

During its consultative review of the “Draft Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center (INTEC) Tank Farm Facility (TFF),” dated September 2005, the U.S. Nuclear Regulatory Commission (NRC) staff made some observations regarding the U.S. Department of Energy’s (DOE) waste management approach. As a result of these observations, the NRC staff has developed some recommendations for DOE’s consideration. The purpose of the recommendations is to communicate actions that DOE might consider to further enhance its approach for management of the TFF waste at INL, as well as the approach for future waste determinations. As stated in this Technical Evaluation Report (TER), the NRC staff has concluded that DOE has adequately demonstrated that it can meet applicable criteria of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA). Thus, it is the NRC staff’s view that implementation of these recommendations is not necessary to meet the criteria in the NDAA. The recommendations are based on the information provided in the TER, and a more detailed discussion of the underlying bases for the recommendations can be found in the referenced sections.

The following recommendations are noted with respect to meeting Criterion Two:

- Using information U.S. Department of Energy, Idaho Operations Office (DOE Idaho) has gained from previous sampling campaigns, DOE Idaho should identify a sampling strategy that has the highest probability of success of obtaining representative samples of the waste (e.g., use of a small submersible pump moved across the bottom of the tank to collect residual heel samples to capture a representative collection of residual materials from a larger portion of the tank) (Section 3.2).

- Using information provided to NRC in response to requests for additional information (RAIs) 2 and 17, DOE Idaho should consider assessing the quality of the solids sample retrieved from tank WM–183 in a revised data quality assessment. DOE should indicate how it has resolved issues that have historically plagued analytical sampling of Tc-99 and I-129 in this assessment (Sections 3.1 and 3.2).

- While NRC has concluded that DOE has met Criterion Two for this waste determination, in any future waste determinations, DOE should establish remedial goals. DOE would need to demonstrate that the remedial goals will result in removal to the maximum extent practical and that the remedial goals will meet the performance objectives (Sections 3.7 and 3.8).

- DOE Idaho should continue to stay abreast of cleaning technologies for potential use in any future activities related to waste determinations (Sections 3.5 and 3.6).

- Use of ORIGEN2 ratios for estimating the solid residual inventory of highly radioactive radionuclides (HRRs) is not recommended for the reasons stated in Section 3.2.1. DOE Idaho should continue to sample HRRs to ensure adequate inventory estimates for the purposes of demonstrating compliance with the NDAA criteria.

The following recommendations are made with respect to Criterion Three:

- DOE Idaho should continue to evaluate and enhance radionuclide release models for grouted systems, which may include assessing (i) the applicability of the K_d approach, (ii) the appropriateness of using cementitious material K_ds for waste that may not be thoroughly mixed with the poured grout, (iii) the need for leaching experiments on expected wasteforms in the potential range of physical and chemical conditions, and (iv) grout pore water chemistry effects on future releases from the Tank Farm Facility (TFF).

- DOE Idaho should consider updating/revising the performance assessment (PA) model to consider recent monitoring data and modeling activities performed for the TFF under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) program (e.g., DOE Idaho, 2004, 2006d,e) to ensure hydrogeological conceptual model uncertainty is appropriately addressed. This would include (i) effects of historical releases on future contaminant transport, with consideration of the work in DOE Idaho (2006e); (ii) additional data to support a hydrology model oriented in the principal vadose zone flow direction (i.e., generally to the southeast) if a two-dimensional model is retained for modeling contaminant flow and transport in the PA; (iii) additional data on Big Lost River seepage rates and underflow rates; and (iv) more realistic parameterization of natural (stratified) alluvium with a representative property set that is separate and distinct from the property set used to model anthropogenically disturbed (homogenized) alluvium (this should provide more realistic, spatially variable infiltration rates). DOE Idaho should consider the effects of infiltration controls and any future cap on contaminant flow and transport at the TFF.

- DOE Idaho should enhance quality assurance controls of documentation in future waste determinations and supporting documentation. This TER and NRC staff’s RAI (NRC, 2006a) note several errors in waste determination, PA documentation, and RAI responses.

- As cleaning and closure of the TFF progresses, the closure strategy for structures, systems, and components related to high-level waste reprocessing, storage, and disposal that may be related to this (or the subject of future) waste determination(s) should be refined based on information obtained during closure of the TFF.

CONTRIBUTORS

U.S. Nuclear Regulatory Commission Staff

Project Manager for INL TFF Determination Review
Xiaosong Yin
Project Manager
M.S., Nuclear Engineering, University of Missouri
M.S., Physics, Miami University

Lead Technical Reviewer
Cynthia S. Barr
Systems Performance Analyst
M.S., Environmental Systems Engineering, Clemson University
B.S., Mathematics, College of Charleston
B.A., Political Science, College of Charleston

Center for Nuclear Waste Regulatory Analyses

Lane D. Howard
Senior Research Engineer
M.S., Nuclear Engineering, Texas A&M University
B.S., Civil Engineering, Texas A&M University

Roberto T. Pabalan
Institute Scientist
Ph.D., Geochemistry and Mineralogy, The Pennsylvania State University
B.S., Geology, University of the Philippines

David A. Pickett
Principal Scientist
Ph.D., Geology, California Institute of Technology
M.S., Geology, California Institute of Technology
B.A., Geology, Rice University

Southwest Research Institute®

Cynthia L. Dinwiddie, Ph.D., P.G.
Senior Research Engineer (Hydrogeologist)
Ph.D., Environmental Engineering and Science, Clemson University
M.S., Environmental Systems Engineering, Clemson University
B.S.E., Walla Walla College

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